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SIMULATION OF A SEVERE ACCIDENT AT A TYPICAL PWR DUE TO BREAK OF A HOT LEG ECCS LINE USING MELCOR CODE

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ABSTRACT

The aim of this work was to simulate a severe accident at a typical PWR caused by break in Emergency Core Cooling System (ECCS) line of a hot leg using the MELCOR code. The nodalization of this typical PWR was elaborated by the Global Research for Safety (GRS) and provided to the CNEN for analysis of the severe accidents at the Angra 2, which is similar to that PWR. Although both of them are not identical the results obtained for that typical PWR may be valuable because of the lack of officially published calculation for Angra 2. Relevant parameters such as pressure, temperature and water level in various control volumes after the break in the hot leg were calculated as well as degree of core degradation and hydrogen concentration in containment. The result obtained in this work could be considered satisfactory in the sense that the physical phenomena reproduced by the simulation were in general very reasonable, and most of the events occurred within acceptable time intervals. However, the uncertainty analysis was not carried out in this work. Furthermore, this scenario could be used as a base for the study of the effectiveness of some preventive or/and mitigating measures of Severe Accident Management (SAMG) by adding associated conditions for each measure in its input.

1. INTRODUCTION

Severe accidents are beyond design basis accidents which involve significant core degradation at nuclear power plant (NPP) [1]. The most recent example of this kind of accident occurred at the Fukushima Daiichi nuclear power station in March 2011.

Since the beginning of the present century Brazilian and international organizations which control the use of nuclear energy, i.e., the CNEN (*Comissão Nacional de Energia Nuclear*) and the IAEA [2, 3, 4], respectively, have been devoting efforts in performing simulations using computer codes for severe accidents at nuclear installations. The aim was to ensure the safety of the surrounding population. The MELCOR [5, 6, 7], the SCDAP/RELAP5 [8, 9] and the MAAP4 [10] are the most widely used simulation codes for this purpose among others.

Since all Brazilian nuclear power plants are pressurized water reactors (PWR), the efforts of the CNEN as well as of the IPEN (*Instituto de Pesquisas Energéticas e Nucleares*) are concentrated on accidents at this type of reactors, specifically on scenarios of SBO (Station Blackout), SBLOCA (Small Break Loss of Coolant Accident) in a cold leg and of

LBLOCA(Large Break Loss of Coolant Accident) in a hot leg, which are considered as the most likely cases of this kind of accident at a PWR. Obviously, other aggravating conditions must be added into these scenarios in order to lead to a beyond design basis accident, such as loss of suction from the sump or/and from the RHR (Residual Heat Removal) system.

The objective of this work is to simulate a basic scenario of LBLOCA which consists of a break of Emergency Core Cooling System (ECCS) line of a hot leg at a typical PWR using MELCOR code. The model of this PWR is developed by the Global Research for Safety (GRS or *Gesellschaft für Anlagen- und Reaktorsicherheit*) in partnership with the CNEN. The same model represents a reactor which is similar to Angra 2, but they are not identical. Fig. 1 and Fig. 2 show the nodalization of the Reactor Cooling System (RCS) and of the containment, respectively, of that PWR.

MELCOR is, as well as MAAP and ASTEC, one of the fast running integral computer codes, which is developed at the Sandia National Laboratory of the USA for the simulation of the relevant phenomena within light water reactor NPPs, either BWR or PWR, in the case of a severe accident [11]. The calculations performed by means of the MELCOR code include a full range of physical phenomena, from thermal-hydraulics to fission product release and transport [5]. The version used in this work is MELCOR 1.86.

Even though the model is relatively simple and the Angra 2 cannot be properly represented by it, the result obtained in this work may be used as a reference for the further numerical simulations of severe accidents at this NPP, even using a more improved nodalization, since, so far, there is no published result of this kind of simulation for it.

2. METHODOLOGY

The simulated scenario is a basic one to which no preventive or mitigating measure was added, and the leak of the LBLOCA is implemented in the control volume of number 200, that is, CV200, of the Loop 1 as shown in Fig. 1. The flow path area of the ECCS injection line at the hot leg is 380cm^2 and its total separation from the pipe of RCS due to the break is postulated. Therefore, at the moment of the break (t = 0 sec) a mass flow is initiated between CV200 and the compartment of the containment which encompass the hot leg, namely, the CV003 as shown in Fig. 2; obviously, the area of this flow path is 380cm^2. The flow paths' number is 445 (FL445) and its area, controlled by a control function CF397 which consists of a tabular function.

The boundary conditions of the present scenario are as follows:

a. Turbine bypass not available;
b. Condenser not available;
c. Loss of suction from the sump and RHR;
d. ECC injection from Refueling Water Storage Tank (RWST) by Safety Injection Pump (SIP) and by RHR pump available; and
e. All accumulators available.

The problem time of the simulation was 12 hours.

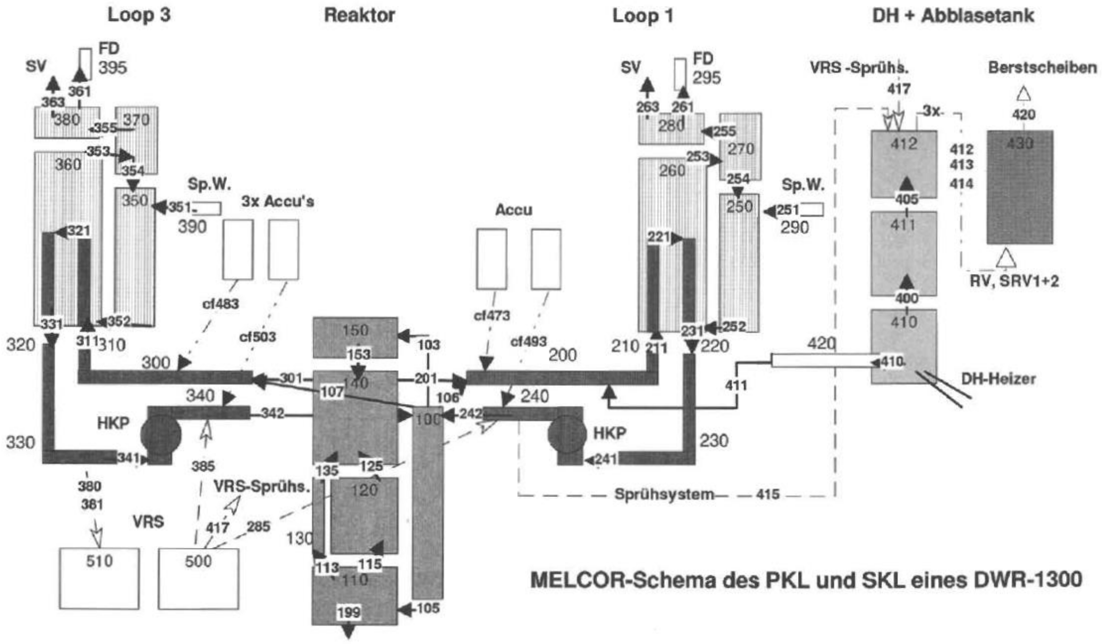

Figure 1: Nodalization of primary and secondary circuit of the PWR.

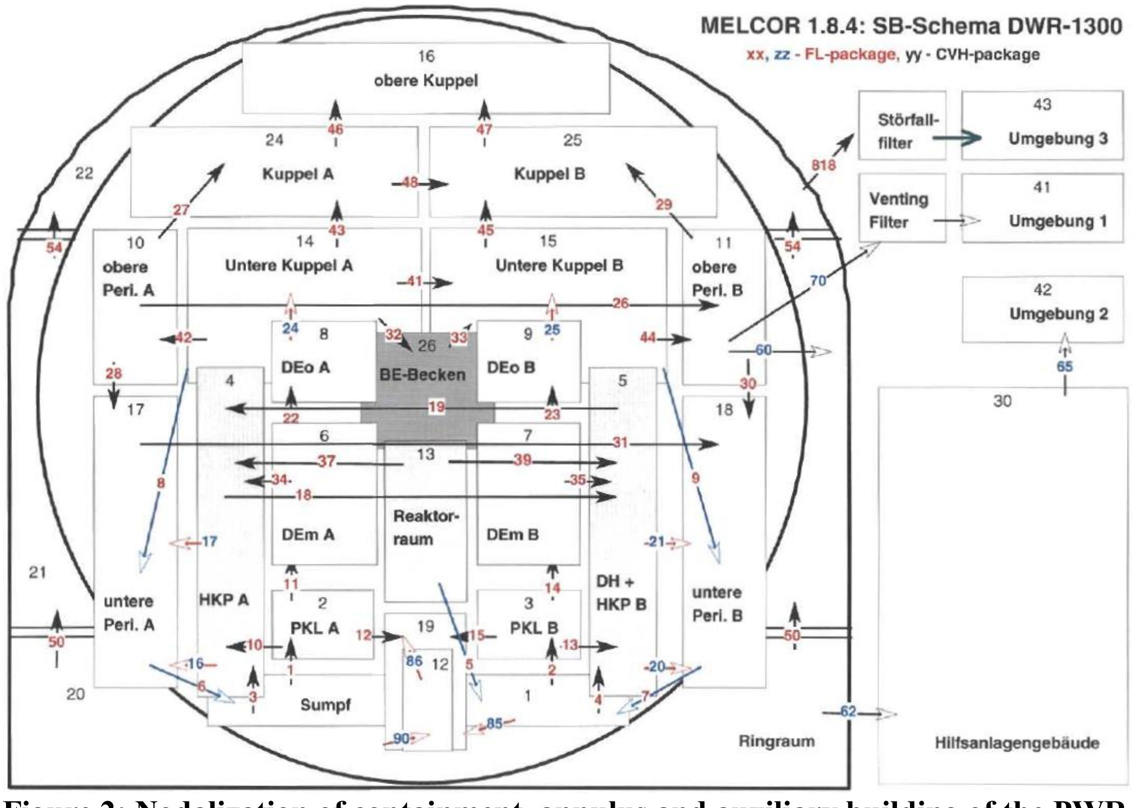

Figure 2: Nodalization of containment, annulus and auxiliary building of the PWR.

3. RESULTS AND DISCUSSIONS

Fig. 3 shows that the degradation of the core starts at 2*h* 48*m* and it is completely melted at 4*h* 23*m*. The consequences of the core melting are shown in Fig. 4, where the hydrogen production due to the oxidation of the metallic structure of the core by steam is presented, as well as the generation of the same element and the carbon monoxide during the Molten Core Concrete Interaction (MCCI). It is worth reminding that no mitigation measure such as Passive Autocatalytic Recombiner (PAR) is implemented yet in the present work.

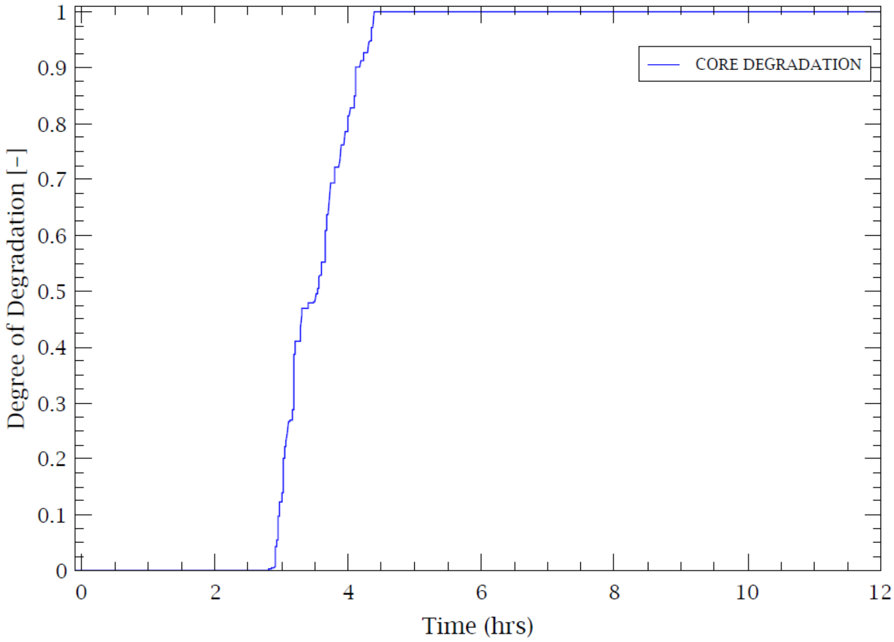

Figure 3: Degree of the core degradation.

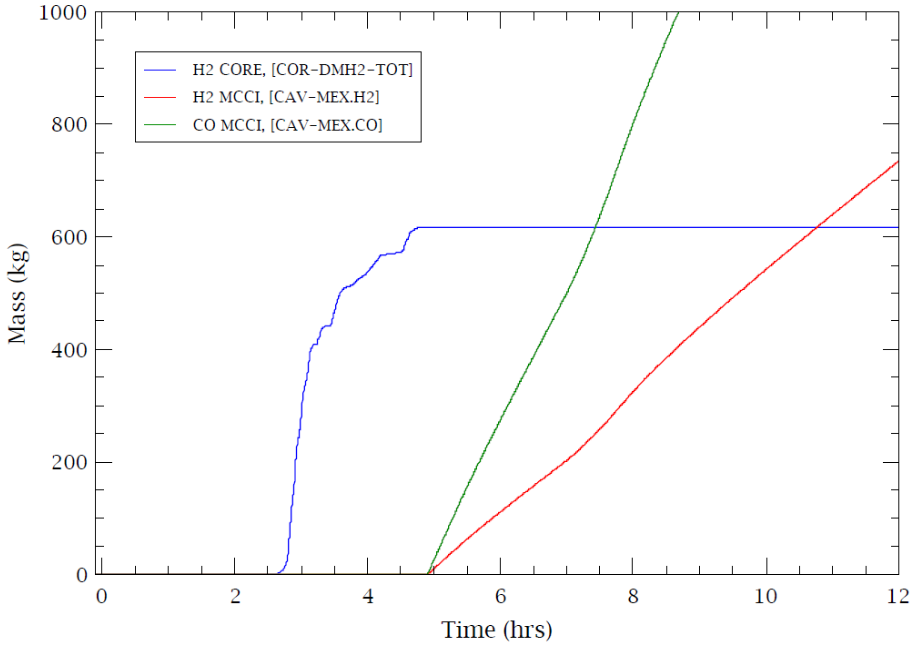

Figure 4: Generated H_2 and CO.

The maximum temperature attained in the core is 2839 ºC as shown in Fig. 5, and the hottest cladding and fuel reach 2226 ºC.

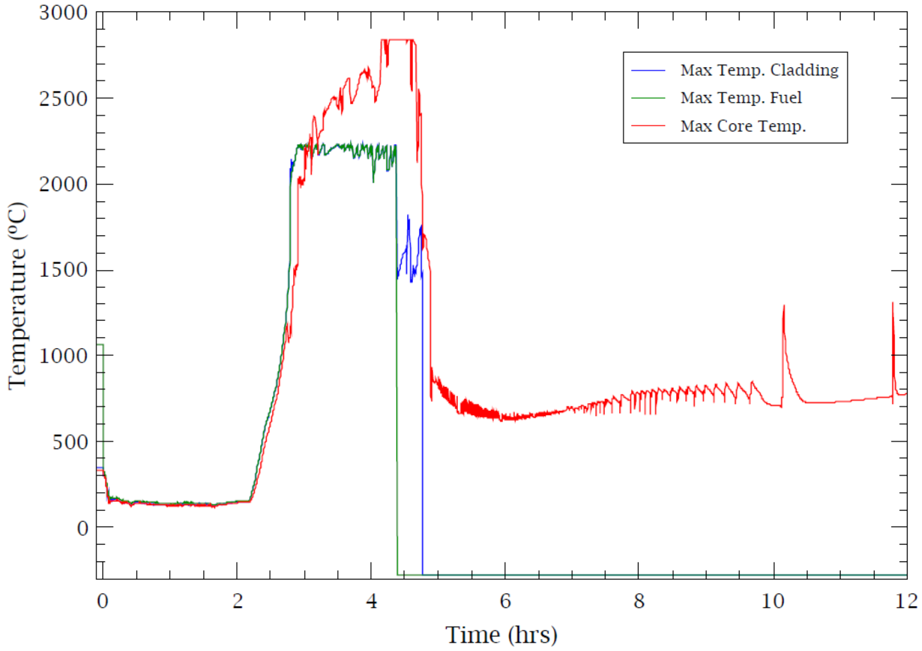

Figure 5: Maximum Temperatures of Cladding, Fuel and the Core.

The pressure in Reactor Pressure Vessel (RPV), Pressurizer, Steam Generators (SG) and Feed Water Tank (FWT) are presented in Fig. 6. Pressures in RPV and Pressurizer drop from about 160 to 97 *bar* within 10 seconds, and then decrease still rapidly to about 3.5 *bar* within 5 *min*. The fast depressurizations reflect the magnitude of the break.

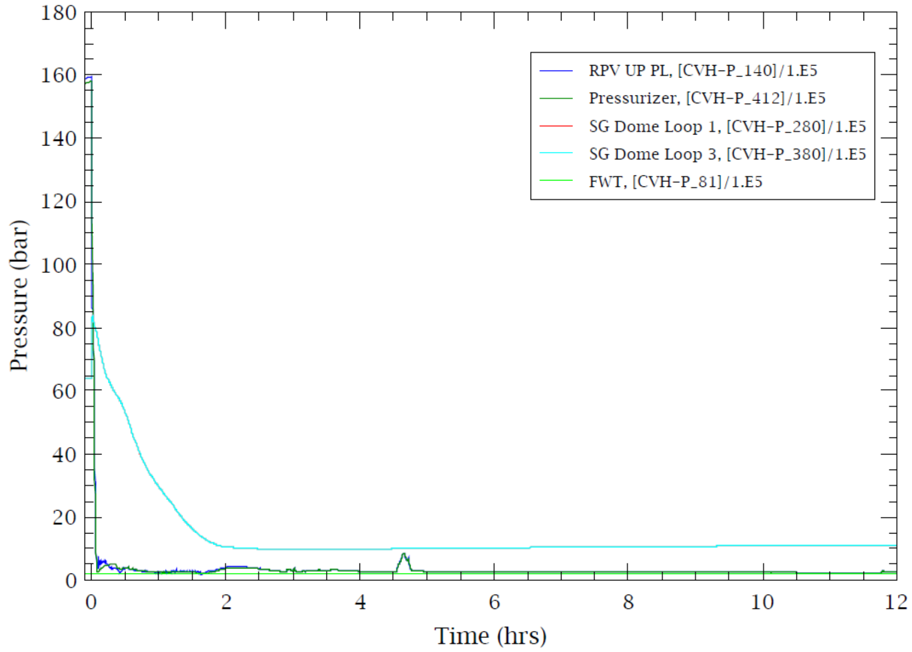

Figure 6: Pressure in RPV, Pressurizer, SG and FWT.

On the other hand, the pressures in the secondary side, that is, in SGs of the both loops increase up from 63 to 84 *bar* in 40 seconds after the break, and immediately after they turn to decrease somewhat slowly to around 10 *bar* at 2 *h*. These depressurizations are due to the reverse heat transfer, from secondary to primary side.

Immediately after the break of the ECCS line, the pressure of the containment jumps to 3.15 *bar* as illustrated in Fig. 7, but it quickly drops down due to the rapid condensation of the leaked steam by the metallic structure inside the containment and, also, due to the ECCS injection. However, after the ECCS injection if finished, the pressure of the containment increases again and the subsequent spike (4 *h* 35*m*) shows that relocation of the melted core into the Lower Plenum (LPL).

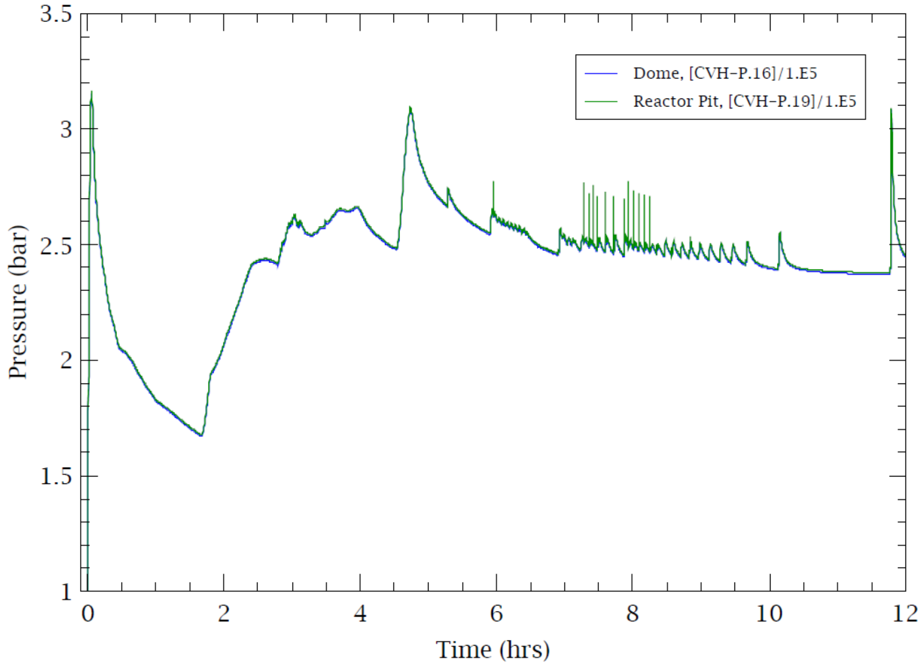

Figure 7: Containment Pressure.

4. CONCLUSIONS

The present simulation reproduced successfully the purposed severe accident, having implemented the postulated boundary conditions and using the available nodalization, since the reactor core was indeed melted down. However, as it was noted in the beginning, this is only a basic and initial scenario unto which further conditions are to be added. These additional conditions could be aggravating, mitigating or even preventive ones depending on each scenario. Also, uncertainty analysis is to be done for the validation of the result. Moreover, some improvement in the nodalization itself will be carried out in order to make it more realistic and/or to make it suitable for a specific NPP such as Angra 2.

In this context, the following steps are proposed:

a) Improvement and subdivision of the nodalization;
b) Implementation of the preventive and mitigating measures of SAMG;
c) Performing of the uncertainty analysis; and
d) Modification of the nodalization in order to fit it to Angra 2.

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