SIMULATION OF A STATION BLACK OUT AT THE ANGRA 2 NPP WITH MELCOR CODE

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Abstract. The interest in evaluating the level of resistance of a nuclear power plant in response to an accident that exceeds the project bases, increased significantly after the Fukushima-Daiichi accident. Melcor is an integrated code, developed by Sandia National Laboratories, used to model and simulate the evolution of severe accidents in nuclear power plants. The Melcor modeling is general and flexible, making use of a "control volume" approach in describing the thermal hydraulic response of the plant. Reactor-specific geometry is imposed only in modeling the reactor core. The reactor cooling circuit and the four SG are represent by two model-loops, a single loop with the pressurizer and an agglutinated triple loop. All active safety systems which depend on AC power are assumed to be unavailable in this analysis. The most important strategies assumed were primary side depressurization and additional makeup water to reactor coolant system. The passive severe accident management measures primary bleed, secondary side bleed, passive injection from feedwater system and firefighting pool available. In Brazil there is the Almirante Álvaro Alberto Nuclear Power Plant that has two plants in operation, and one of them is Angra 2, which started operating in 2001. This unit is a pressurized water reactor type with electrical output of about 1350 MW. The objective of this work is to present a summary of the severe accident caused by a station black out condition using the Melcor 1.8.6 code. The main result of the study is an evaluation of RPV lower head integrity during a severe accidents scenario. The results will be useful to an independent assessment into the detailed processes involved by the management guidelines for one scenario severe accident in Angra 2.

Keywords: Severe Accident, Station Black Out, Melcor, PWR, Mitigation

1. INTRODUCTION

The Brazilian National Nuclear Energy Commission (CNEN) is responsible for the regulation of nuclear safety and the promotion, orientation and coordination of nuclear research and technological development.

After the March 2011 Fukushima nuclear accident, CNEN required the development and implementation of a Severe Accident Management Programs (SAMP) for CNAAA - Almirante Álvaro Alberto Nuclear Centre, unit 2 (Angra 2) following guide IAEA SRS 32 [1]. The Eletronuclear (ETN) is a government entity responsible for nuclear power plants operation. To meet this requirement, the ETN prepared the Action Plan 2PA-001.2011 [2] and as part of this plan was develop a SAMP to Angra 2. ETN also submitted to CNEN a deterministic safety analysis with the Melcor code

As CNEN practice, an independent analysis has been carried out by the regulatory body to auxiliary in the assessment process of the safety analysis report presented by ETN. The Melcor version used for this analysis is the same utilized by ETN.

This analysis was development supported by the BR3.01/12 project [3] of the INSC Programe. The objectives of this project is support to the Nuclear Safety Regulator of Brazil. The overall objective of the project is part of the enhancement d strengthening of a nuclear safety regulatory regime in Brazil in compliance with internationally used criteria and practices. One of the purposes of this project is to support CNEN to develop a own MELCOR nodalisation for the Angra 2 NPP.

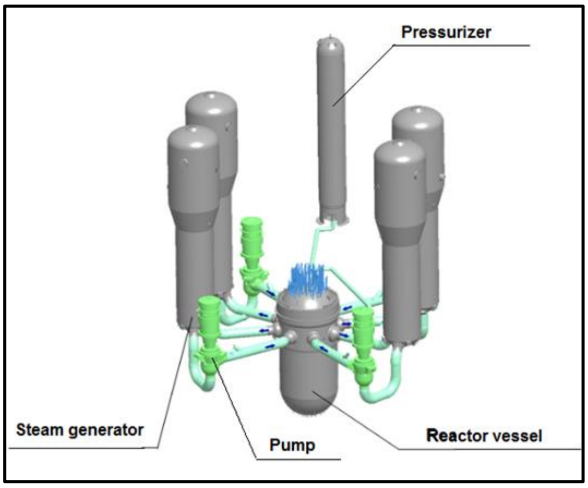
2. DESCRIPTON OF THE ANGRA 2 DESIGN

Angra 2 is the second Brazilian Nuclear Power Plant (NPP) located at the CNAAA on the Itaorna beach in Angra dos Reis, Rio de Janeiro, Brazil. It achieved full power operation in 2001.

This plant has a PWR built by Siemens-KWU (currently Areva NPP), resulting from an agreement between Brazil and Germany in 1975. Angra 2 is a reactor with 1,350MWe capacity providing energy to a 2-million-inhabitant city.

Designed as a four-loop plant, the PWR is based on a proven-technology of other four-loop plants. Angra 2 has four main coolant pumps to control water flow of each loop. To guarantee the safety of this nuclear power plant two Emergency Core Cooling System (ECCS) with four trains is existing. For each loop one ECCS train is connected (one hot and one cold ECCS injection) [4].

Figure 1 shows the arrangement of the components of Angra 2 NPP and Figure 2 shows the view of Angra Nuclear Power Plant Site.



Source: EJAM, Vol. 5, No. 1, NT54, 2009 (Mitsubishi Heavy Industries, 2009).

FIGURE 1. General arrangement of Angra 2 nuclear power plant components

FIGURE 2. View of Angra Nuclear Power Plant Site

The major components of the Reactor Pressure Vessel (RPV) are the reactor vessel, the core barrel, the reactor core and the upper internals package, as can show in Figure 3.

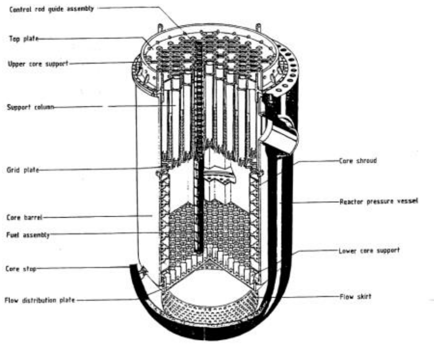


FIGURE 3. Reactor Pressure Vessel

The steam generator (SG) is a vertically arranged U-tube bundle heat exchanger.

The most important components of the primary side are:

- a horizontal tube sheet into which the U-tube are set. The tube are flared on both sides of the sheet and are welded;
- A hemispherical plenum below the tube sheet. The plenum is separated into two chambers.

The Figure 4 shows SG components.

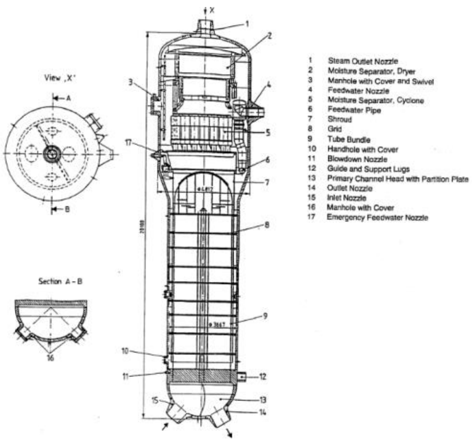


FIGURE 4 . Steam Generator

Measures taken by Angra 2 NPP which are related to severe accident management in order to meet the Brazilian Regulatory Body (CNEN) requirements. These requirements areassociated with the Stress Tests results and were defined after Fukushima severe accident:

a) Prepared Action Plan 2PA-001.2011, which follows an approach similar to the one adopted for the German NPPs of the Angra 2 similar design:

- Development of Severe Acident Management Guidelines (SAMG) based on the German concept;
- SAMG incorporates additional equipment, dedicated for control and mitigation of Severe Accidents;
- Incorporation at a later time of findings from ETN Fukushima Response Plan [2]

b) Planned and installed Plant modifications associated with the Angra 2 SAMP:

- Complementing pressurizer valve station to allow Bleed and Feed (B&F) through th Relief and Safety valves;
- Passive autocathalytic recombiners (PARs);
- Filtered Containment Venting;
- Containment Sampling System for Severe Accident conditions;

Additional mobile equipment: Small Emergency Diesel Generators, Diesel driven pumps (Fukushima response Plan-Angra 2 Stress Test [5]).

3. MELCOR ANGRA 2 NODALIZATION

Melcor is a severe accident code developed by Sandia National Laboratories for the NRC. Its primary purpose is to simulate the evolution of accidents in light water nuclear reactors and to generate fission product source terms. MELCOR is composed of several different modules, called packages (which are fully integrated), that model the important phenomena that can occur during severe nuclear accidents.

The Angra 2 Melcor accident management model was built to simulate various types of transient and accident scenarios, involving complete or partial failure of plant systems and their components. In this work are being presents the SBO scenario accident accident and the Angra 2 response for this hypothetical condition scenario.

The Angra 2 Melcor plant model was developed consist of:

- The reactor coolant system;
- The reactor core;
- The reactor cavity;
- The steam generators and secondary system;
- The containment and the reactor annulus
- The nuclear auxiliary building;
- The emergency core cooling and the emergency feed water systems.

More detailed information about the model is given in next itnes.

3.1 Reactor circuit and steam generators

The reactor cooling circuit and the four SG are represent by two model-loops, a single loop with the pressurizer and an agglutinated triple loop. Each model loop has five control volumes to the rector cooling system and four control volumes to the SG secondary side: Each loop consists of the HL, SG primary side (U tubes), the pump and the CL. All the U tubes are modeled with two equivalent hydraulic regions. One region represents the ascending U tubes side which is coupled with the riser of the correspondent secondary side by a heat structure. The other represents the U tubes descending side and also coupled with the riser of the correspondent secondary side by another heat structure, as shown in Figure 5 and in the Table 1 the control volumes are identify.

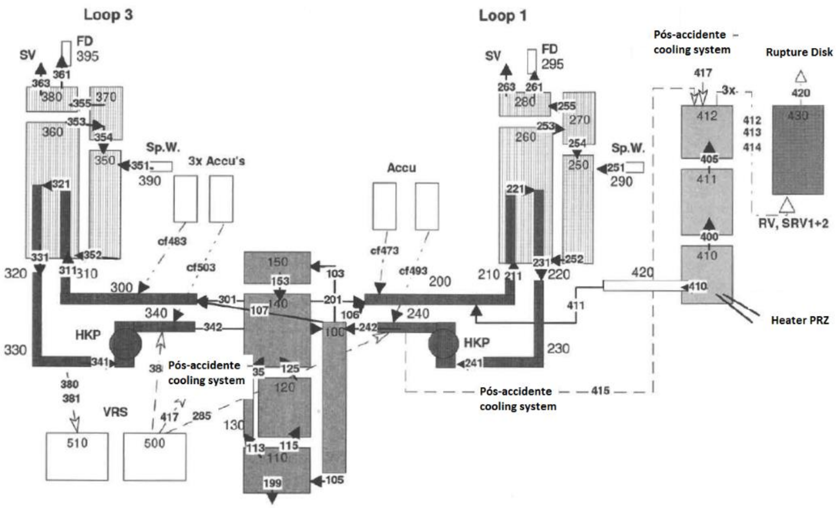


FIGURE 5. Reactor circuit and steam generators nodalization

TABLE 1. Reactor Circuit and Steam Generator Control Volumes

CONTROL VOLUME	DESCRIPITION
CV200/CV300	Hot leg including SG entrance chamber
CV210/CV310	Rising SG tubes
CV220/CV320	Failing SG tubes
CV230/CV330	SG exit chamber and cold leg to RCP
CV240/CV340	Cold leg from RCP to RPV
CV250/CV350	SG downcomer with the MFW
CV260/C360	SG riser
CV270/CV370	SG downcomer upper part
CV280/CV380	SG steam dome

3.2. Reactor Pressure Vessel Nodalization

The core and lower-plenum regions of the RPV are divided into 5 concentric radial rings and 15 axial. The levels with the 4 to 14 range axial levels representing active core. Since in-vessel retention modeling capabilities of the code are of particular interest here, the lower-plenum region was divided into 6 radial rings together with 2 axial levels. The Figure 6 shows the RPV nodalization.

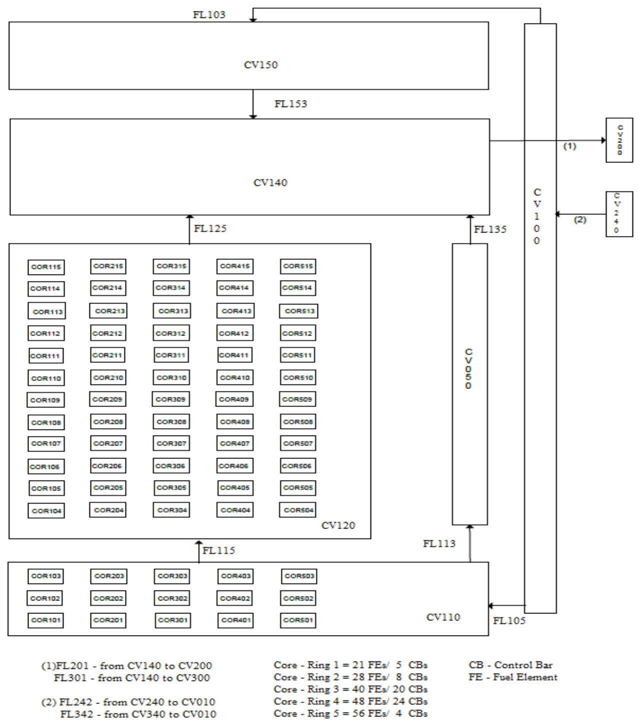


FIGURE 6. Reactor Pressure Vessel Nodalization

3,3.Passive injection to supply the Secondary Feed

Under the Design Base Extension Condition (DBEC), during secondary bleed and feed, water injection into the steam generator can be achieved by using two different methods: Passive injection from Feedwater Tank Inventory and from Fire Fighting Water Inventory.

3.3.1 Passive injection from Feedwater Tank Inventory

The SG feed can be maintained using the feedwater piping and tank inventory for approximadely 4 h [4] if:

- the level in the feedwater tank is in the normal range.
- the pressure in the feedwater tank is appros. 4.9 bar or the feedwater tank can be pressurized to this pressure.
- the SG can be depressurized beforehand to approximately at 1 – 2 bar.
- the residual steam inventory of all four dried out SG is sufficient to pressurize the feedwater tank from 2 to 4.9 bar.

Figure 7 shows the Passive Injection from Feedwater Tank Inventory and Table 2 presents the its control volumes.

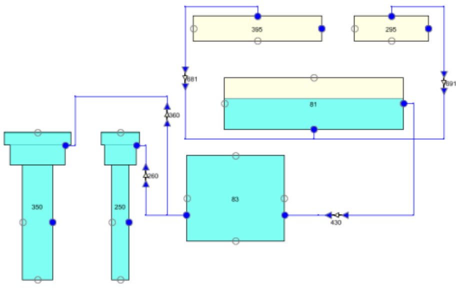


FIGURE 7. Passive Injection from Feedwater Tank Inventory

TABLE 2. Description of the Passive Injection from EFWS components

Control Volume Nº	Volume(m^3)	Description
81	600.0	Feed Water Tank
83	300.0	Feed Water Line
250	50.0	SG Downcomer A
295	50.0	SG1 main SL
390	150.0	SG Downcomer B
395	150.0	SG3 main SL

3.3.2.Passive injection from Fire Fighting Water Inventory modeling

The SG feed can be maintained using the fire water fighting tank for approx. 27 h (residual heat removal and removal of plant stored heat) [4].

The Bernoulli equation is the approach applied to modelling the passive injection from Fire Fighting Water Tank. This system is illustrated by the Figure 8.

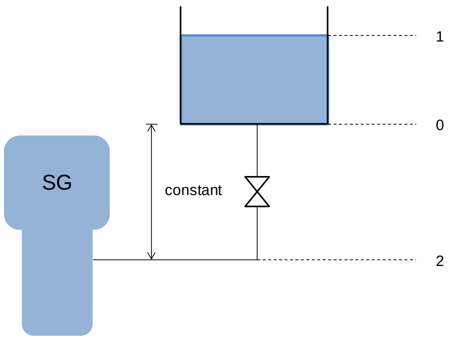


FIGURE 8. Passive Injection from Fire Fighting water Inventory Scheme

Applying Bernoulli equation 1, could be calculated the flow rate in the SG downcomer, as follow demonstrated.

$$P_1 + \rho g h_1 + \rho \frac{w_1^2}{2} = P_2 + \rho g h_2 + \rho \frac{w_2^2}{2} + \Delta P_t \ \ (\div \rho)$$

$$\frac{w_1^2}{2} \approx 0$$

$$\frac{P_1}{\rho} + g h_1 = \frac{P_2}{\rho} + g h_2 + \frac{w_2^2}{2} + \frac{\Delta P_t}{\rho}$$

$$\frac{w_2^2}{2} = \frac{1}{\rho}(P_1 - P_2) + g\,(h_1 - h_2) - \frac{\Delta P_t}{\rho}$$

$$w_2^2 = \frac{2}{\rho}(P_1 - P_2) + 2g\,(h_1 - h_2) - \frac{2\Delta P_t}{\rho}$$

EQUATION 1. Bernoully equation

The SG feed can be maintained using the fire water pond inventory for approx.. 27 h (residual heat removal and removal of plant stored heat).

Up to an main steam pressure of approx.. 7 bar in the steam generator a sufficient feed rate can be maintained solely because of the elevation head between the fire water pond and the SGs. At higher main steam pressures a mobile pump is required.

3.4.Containment nodalization

The containment is simulated by 23 control volume, the annulus by 3 control volumes. The control volumes are interconnected by flow paths (Flnnn).

The Figure 9 shows the containment nodalization and the Table 3 describe each control volume.

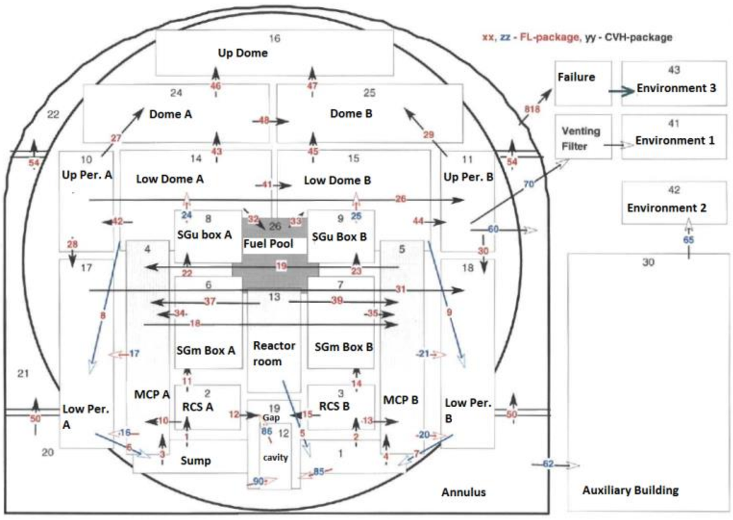


FIGURE 9. Containment Nodalization

TABLE 3. .Containment Control Volume Description

Control Volume		**Description**
Number	Volume (m^3)	
1	3858.0	Sump
2	500.0	RCS A
3	500.0	RCS B
4	6096.0	MCP A
5	5994.0	Prz. MCP B
6	500.0	SGm Box A
7	500.0	SGm Box B
8	270.0	SGu Box A
9	270.0	SGu Box B
10	6919.0	Up Per. A
11	6885.0	Up Per. B
12	143.0	Cavity
13	640.0	RPV Head
14	7038.0	Low Dome A
15	7038.0	Low Dome B
16	2940.0	Up Dome
17	3889.0	Low Per. A
18	3989.0	Low Per. B
19	147.0	Gap
24	5526.0	Dome A
25	5526.0	Dome B
26	1665.0	Fuel Pool

3.5.Passive Autocatalytic Recombiner System

Concerning the PARs, some records are dedicated to calculate the recombination rate of hydrogen or/and carbon monoxide in each control volume (CV) by mean of the following empirical correlation [6] shows by the Equaion 2:

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

EQUATION 2. PAR empirical correlation

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 efficiency,

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, depending on recombiner type [mol/s bar].

There are 60 PARs of the AREVA type installed in Angra 2 and their positions can be seen in the Figure 10 and Table 4 describe each control volume.

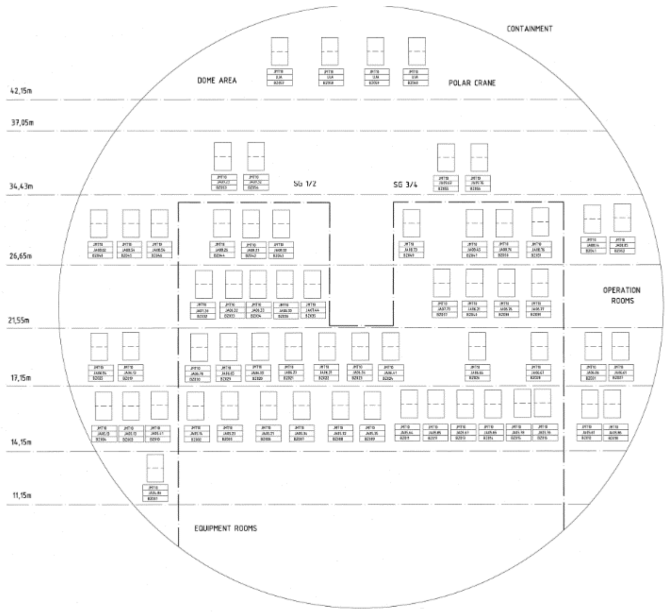


FIGURE 10. Hydrogen Reduction System – PARs position in the containment

TABLE 4. Type, Number and Position of Pars

CV	1500T	1500R	960	380T	320	
4	1		1	1		
5	1			1		
6	4		1	1	3	
7	3		2		2	
8			2	1	1	
9	1			1	1	
10	2		1			
11	2					
13	1					
14	4			1		
15	5			1		
17			3		3	
18	1		2	1	1	
24			2			
25	2					
soma	27	0	14	8	11	60

4. Summary of results of SBO case severe accident scenario in Angra 2 NPP

The following assumptions were defined to simulate SBO, considering that no Reactor Coolant System despressurization was available:

a) Loss of all AC power

b) All accumulators available;

c) No PBF available;

d) SBF available.

The Angra 2 NPP response to this SBO scenario during a short term is presented below.

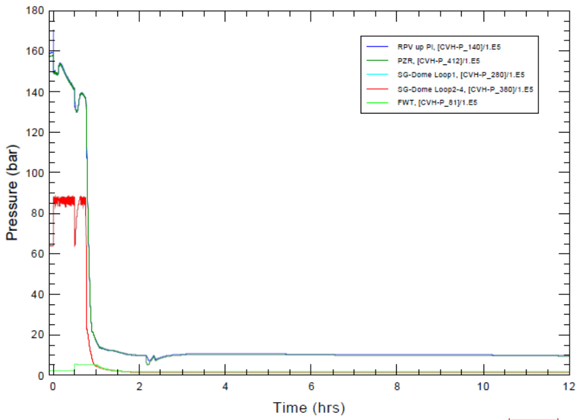


FIGURE 11. Pressure in RPV, PZR, SG and FW Tank

Comentado [AP1]:

Comentado [AP2R1]:

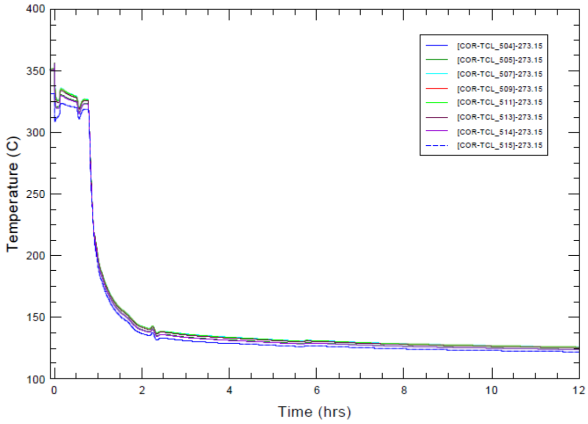


FIGURE 12. Cladding temperature for outermost ring

Under the BDBA assumption, during secondary side bleed and feed, water injection into the steam generator (SG) can be achieved at the Angra 2 plant by using two different methods [4]. Passive injection from Feedwater Tank Inventory, show in Figures 12 and from fire fighting water inventory, as can see in Figure 14.

The SG feed can be maintained using the fire fighting fire water inventory for more then 50 50 h (for the residual heat removal and removal of plant stored heat), when it empties, as can see in Figure 14.

Up to a main steam pressure of approximately 7 bar in the steam generator, a sufficient feed rate can be maintained solely because of the elevation head between the fire water inventory and the SGs, see figure 6. At higher main steam pressures a mobile pump is required.

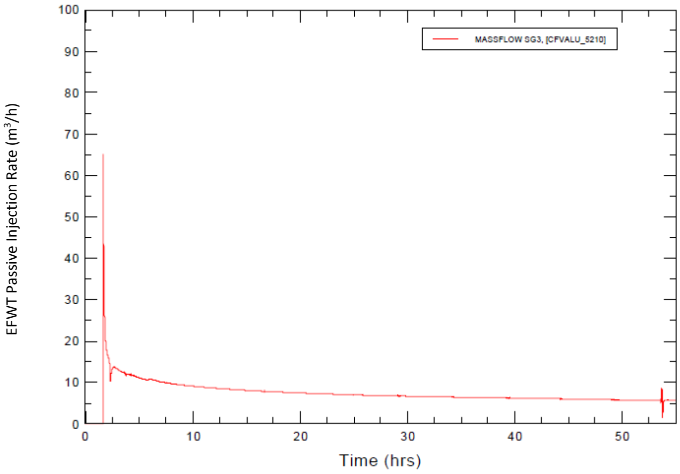


FIGURE 12. Masflow SG1 (single loop), passive injection from the EFWT

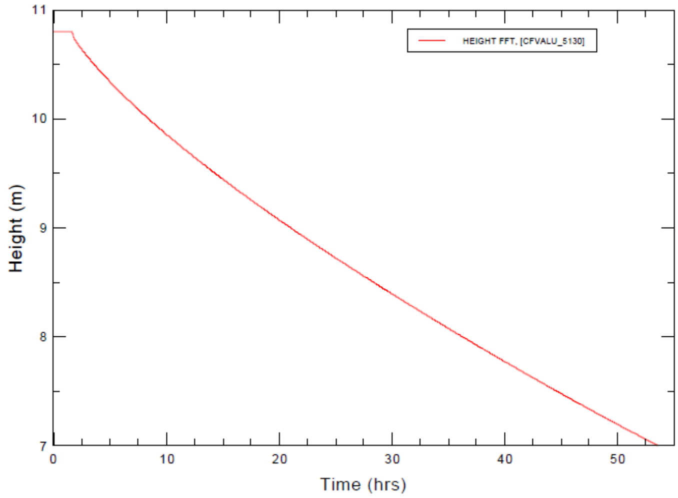


FIGURE 13. Height in the Fire Fighting Tank

Due to the extensive time during which the reactor core is protect by the secondary bleed and feed, more than 50 h, no core damage can be observed in Figures 3 and 4 because the event is simulated only for 12 h.

Figure 14 shows the water level in some components. As expected, the core level do not decrease.

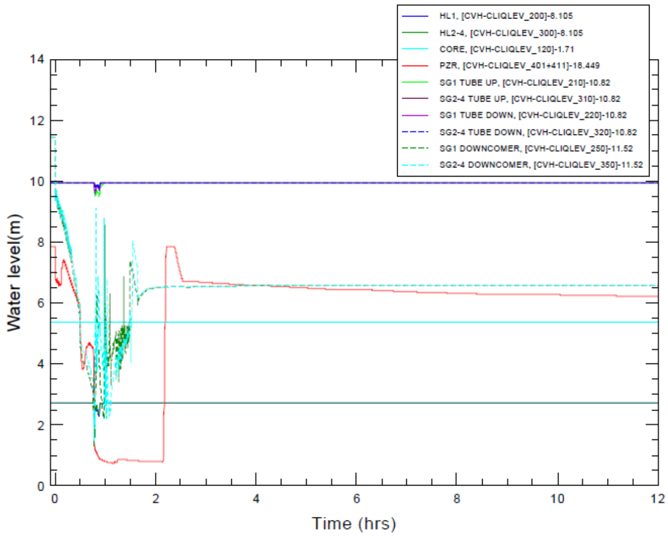


FIGURE 14. Water level in RCS, PZR and SG

The behavior of the containment atmosphere can be seen in the Figures 15 and 16.

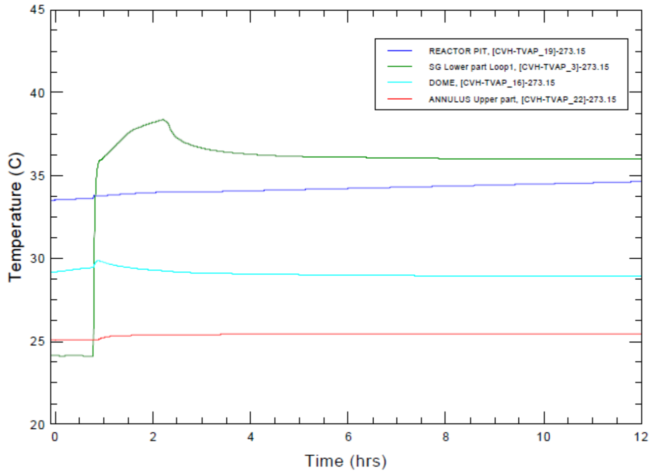


FIGURE 15. Containment atmosphere temperature (short-term)

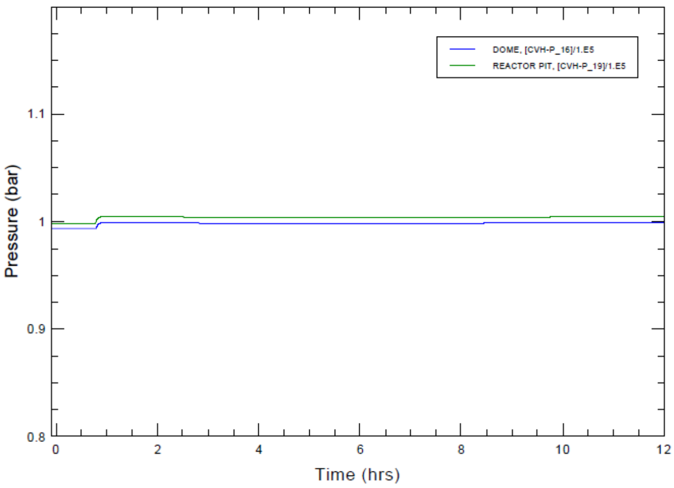


FIGURE 16. Containment pressure (short-term)

The results obtained through the simulation of the SBO scenario as presented show that the core damage does not occur in the first 12 hours after the onset of the event. Thus, there is, at least, a period of 12 h to take the necessary steps to leave the plant to a safe mode.

3. CONCLUSION

Based in the results obtained in this analysis we can conclude that Angra 2 NPP is resistant against a SBO scenario of severe accident in the first 12 h after the event begin.

The importance of plant-specific severe accident simulations has been shown. This should however never replace engineering judgement and experience, since they are two complementary aspects for a state-of-the-art SAMG implementation.

NOMENCLATURE

CNEN- Brazilian National Nuclear Energy Commission

CNAAA - Almirante Álvaro Alberto Nuclear Centre

CNEN – Brazilian Regulatory Body

ECCS - Emergency Core Cooling System

EFWS – Emergency Feedwater System

INSC - Instrument for Nuclear Safety Cooperation

NNEC - National Nuclear Energy Commission

NPP – Nuclear Power Plant

PAR – Passive Autocatalytic Recombiner

PWR - Pressurized Water Reactor

RCS – Reactor Cooling System

SA – Severe Accident

SBO – Station Black Out

SAMG – Severe Accident Management Guide

SAMP – Severe Accident Management Program

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Acknowledgments

We would like to acknowledge the European Union, who sponsored and provided the needs for the development of BR3.01/12 tasks.