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CONTAINMENT PRESSURE ANALYSIS METHODOLOGY DURINGA LBLOCA WITH COCOSYS CODE

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ABSTRACT

During a nuclear power plant basic design accident, the containment integrity is a determining factor for the accident severity. The pressure and temperature conditions inside the containment in case of a Large Break Loss of Coolant Accident (LBLOCA) must be verified. This paper presents a containment pressure and temperature analysis methodology of a Brazilian PWR, Angra 2, using a code that simulates guillotine rupture - RELAP5 - and the COCOSYS code, which analyzes the containment pressure from the accident conditions. The Angra 2 containment behavior results during the design basis accidents studied - primary cooling system cold and hot legs guillotine ruptures - were satisfactory when compared to those presented in the Final Safety Analysis Report (FSAR / A2) and the pressure distributions were below the containment design pressure value (6.3bar).

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INTRODUCTION

The International Atomic Energy Agency (IAEA) defined that "the design basis accidents relevant for the design of the containment systems should be those accidents having the potential to cause excessive mechanical loads on the containment structure and/or containment systems, or to jeopardize the capability of the containment structure and/or containment systems to limit the dispersion of radioactive substances to the environment." (IAEA, 2004). One of those accidents is the Loss of Coolant Accident (LOCA), defined as an accident that results in the loss of coolant that goes beyond the restoration capacity of the volumetric refrigeration control system (USNRC, 2017).

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It's a requirement of the nuclear plant design that the containment building supports the pressures and temperatures resulted from this type of event (USNRC, 2017). Thus, Safety Analysis Report of any nuclear facility defined theoretical accident studies simulated with computer codes. In evaluations of this type of accident, computer codes and methods selected to verify the consequences of an initiating event (postulate) must provide enough safety margin[1] for the entire sequence of events within the limits established by the regulatory bodies (IAEA, 2004). All evaluations should be adequately documented with an indication of the analyzed parameters, the adopted computer codes and the acceptance criteria used. In the early '80s, the ability of advanced computational codes to predict behavior during a LOCA evolved. With that, even the conservatism defined at Appendix K of the 10 Code of Federal

[1] The margins considered are related to physical uncertainties, design uncertainties (such as structures) and operating margins (including operator failure).

Regulations (CFR) 50.46 could not be estimated quantitatively. Thus, the U.S. Nuclear Regulatory Commission (USNRC) has adopted a provisional approach to Appendix K assessment models, which are still requirements, but which allow the use of Best Estimate (BE) methods (Galetti, 2007). There are different calculation options of accidents analyses when combining the use of computer codes and input data for licensing purposes. The one used in this study is the conservative-realistic approach (Fiori, 2009), which follows Appendix K in the case of a LOCA, except that Best Estimate computational codes are used instead of conservative codes.

MATERIALS AND METHODS

The methodology used for the containment pressure analysis is presented in the flowchart below (Fig. 1). The COCOSYS V2.4 code was used to analyze the conditions in the containment of the Angra 2 reactor during a LBLOCA. As boundary conditions, the results of a simulation of this same accident were used, calculated by the RELAP5/ MOD3.2 Gamma. This process was repeated more than once (iterative process) and then the containment pressure distribution was analyzed for each iteration. As indicated in a study [5], the conditions in the reactor core are more realistic when containment condition is considered. Although not analyzed in this paper, the results of the core conditions would possibly improve when considering the iterative methodology between RELAP5 and COCOSYS codes.

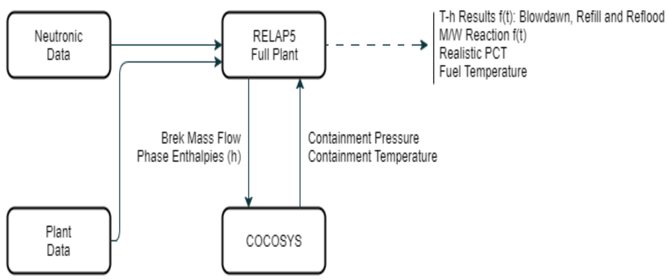

Figure 1. Methodology used in the analysis of the containment with RELAP5 and COCOSYS codes

Plant Description: The Almirante Álvaro Alberto Nuclear Power Plant - Unit 2, located in the state of Rio de Janeiro, is a PWR designed by German Siemens/KWU and operated by Eletronuclear. In a remote case of radioactive material release, the reactor, the primary circuit and the storage pools of fuel elements are surrounded by the containment, which is a WSTE 51 austenitic steel sphere, with internal diameter of 56 m, thickness of 30 mm and mass of 2,600 ton. This structure is protected and surrounded by the secondary containment: a concrete building of cylindrical shape and a concrete dome, with diameter of 60m, thickness of 60cm and height of 60m (Eletronuclear, 2010). The geometric and operational conditions of the Angra 2 containment considered, according to its Final Safety Analysis Report (FSAR/A2) (Eletronuclear, 2010), are listed in Table 1.

Table 1. Numerical results to the model problem

Item	Unit	Value
Internal diameter	m	56,0
Design free volume	m³	$7.1x10^4$
Steel containment thickness	mm	30.0
Design manometric pressure	bar	5,3
Steel Containment Surface	m²	7.66×10^3

If the design pressure of 5.3 bar[2] is reached, the containment relief valve is partially opened at 5% of its total area, so, part of the containment pressure is released to the environment. However, if the containment pressure continues to increase to the maximum 8.5 bar[3], the relief valve will be fully opened - 100% of its area - releasing to the environment not only the pressure but also, in a controlled manner, the waste from nuclear fission occurring in the reactor (Eletronuclear, 2010). The LBLOCA realistic methodology has two additional sensitivity calculations defined in chapter 15.6.4 of FSAR/A2 to determine the containment's highest and lowest pressure, in addition to the base case, which considers the containment's design free volume for each break. These are differentiated by the containment's free volume considered. When considering an internal volume greater than the base case, the internal pressure peak tends to be lower. This approach is called Low Case (volume greater than 5% compared to Base Case). For the High Case, the internal volume is 5% lower than the Base Case, and the pressure values for the same accident are expected to be larger. The TAB. 2 indicates the values of free volumes considered in each of the approaches according to FSAR/A2.

Table 2. Volume for each approach used for LBLOCA simulations

Approach	Containment Volume (m³)
High Case	67.840
Base Case	70.980
Low Case	74.275

Accident Description

The three accidents considered are:

- the rupture of the primary circuit hot leg, between the outlet of the pressure vessel and the input of the steam generator circuit 20 (LBLOCA-HL);
- the rupture of the primary circuit cross-over leg, between the steam generator circuit 20 and reactor coolant pump (LBLOCA-crossover);
- the rupture of the primary circuit cold leg, between reactor coolant pump and de input of the pressure vessel (LBLOCA-CL).

These accidents are described in item 15.6.4.2[4] of the accident analysis chapter of FSAR/A2.

To obtain the containment pressure and temperature in these events, the LBLOCA simulation results of the Technical Report (Sabundjian, 2016) was considered as initial condition, which uses the basic input and nodalization developed by The CNEN Working Group [8] for the simulation of each accident. A nodalization was done for all four coolant loops of the primary circuits, but only one loop (20), which contains the pressurizer, is presented in Fig. 2. This is the loop which the rupture was considered in this work, since it represents the worst scenario in LBLOCA, due to the faster drainage of the surge line and the pressurizer. The initial and boundary conditions adopted in this simulation follow those specified in Table 3.

[2] Manometric pressure of 5,3 bar and absolute pressure of 6,3 bar.
[3] Manometric pressure of 8,5 bar and absolute pressure of 9,5 bar.
[4] Denominaded *Large-Break Loss-of-Coolant Accidents*

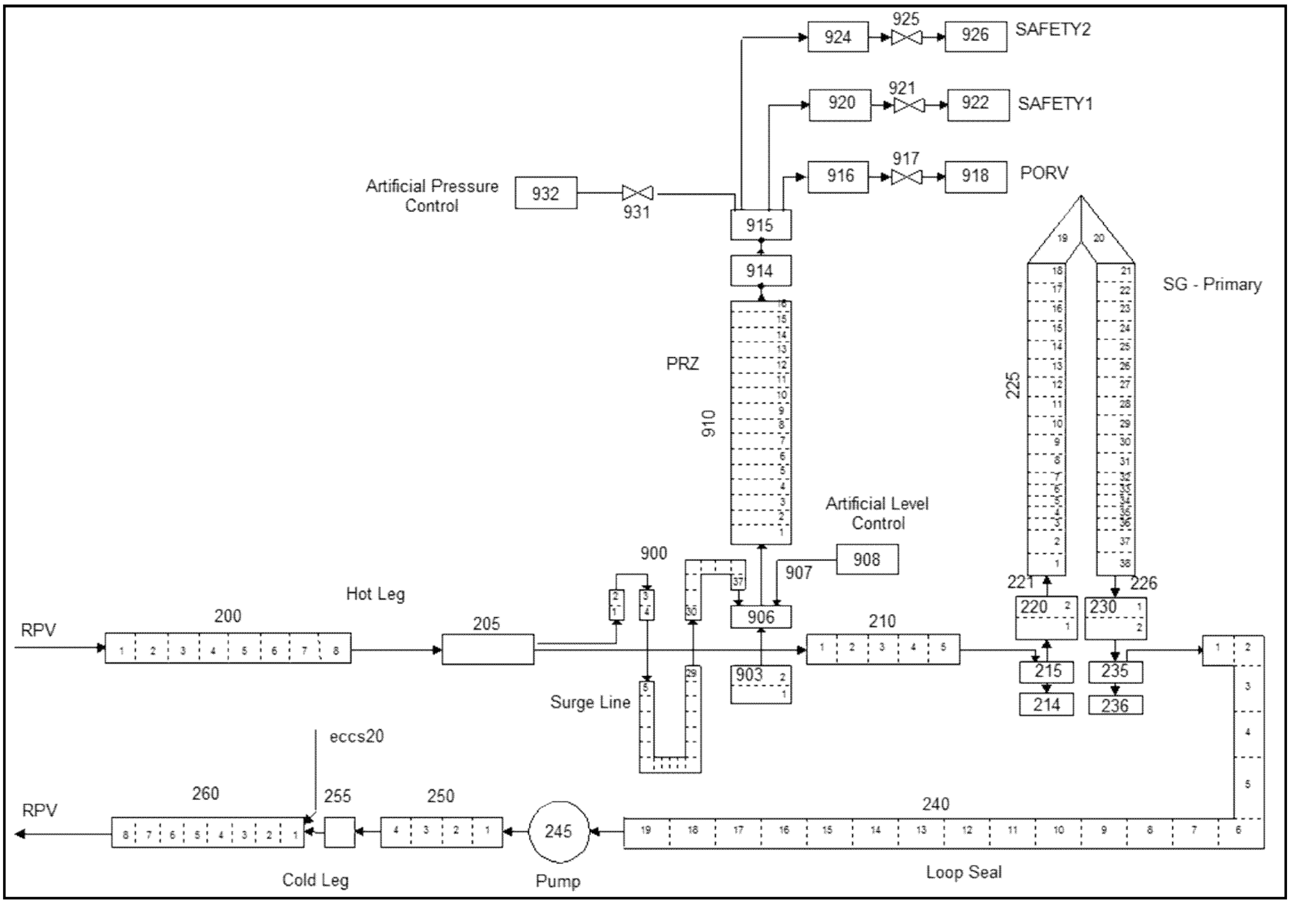

Figure 2. Angra 2 loop 20 nodalization

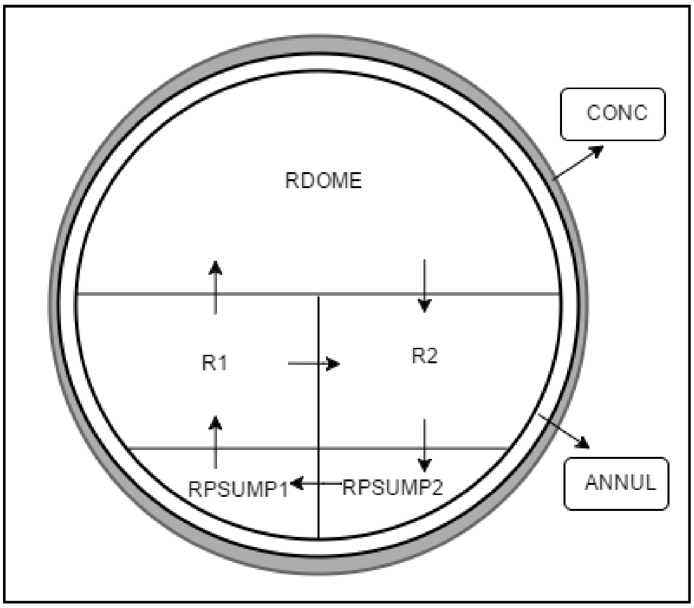

Figure 3. Angra 2 Containment nodalization

Conservative approaches were chosen by assuming restrictive availability of the ECCS with repairs and single failure affecting important components, as listed in Table 4, corresponding to FSAR/A2 Table 15.6.4.2-9 (Eletronuclear, 2010). The accident was simulated with the RELAP5/MOD3.2Gamma code (RELAP5/MOD3 Code Manual, 1999). This code can simulate a LOCA of small, medium or large rupture. In addition, it can simulate transients as loss of electrical power, loss of feed water, loss of flow, among others. The thermohydraulic behavior analysis during one of these accidents or transients con be applied for both systems (primary and secondary). This is a RELAP5 code Best Estimated version. One of the contributing factors is their discharge rate model, which allows to adopt the Henry Fauske model (Henry, 1971). Studies (USNRC, 1982) indicate that this model was less conservative than the Ransom-Trapp model (Trapp, 1992) and Moody model (Moody, 1965) (the one suggested in Appendix K). The FSAR/A2 uses the S-RELAP5 code. That version incorporates features of the RELAP5/MOD2 and RELAP5/MOD3 versions, with some specific improvements adopted by Siemens/KWU (Eletronuclear, 2010).

Table 3. Initial conditions of the Angra 2

Parameter	Unit	Nominal [RFAS/A2]	Relap5/ Mod 3.2gama	Error (%) [1]	
				CALCULATED	ACCEPTABLE
		Reactor			
Thermal power	MW	3765	3768.4	0.09	2.0
Vessel loss of pressure	bar	2.93	2.815	-3.92	10
Core loss of pressure	bar	1.34	1.345	0.37	10
Core outlet temperature	K	601.25	601.18	-0.01	0.5
Core inlet temperature	K	564.45	566.29	0.33	0.5
Core temperature increase	K	36.80	34.89	-5.19	-
Vessel outlet temperature	K	599.25	600.70	0.24	0.5
Vessel inlet temperature	K	564.45	566.29	0.33	0.5
Vessel temperature increase	K	34.8	34.41	-1.12	-
Core coolant flow	kg/s	17672.0	17671.00	-0.01	2.0
Core bypass flow	kg/s	846.00	845.69	-0.04	10.0
Cold-Leg bypass flow	kg/s	188.00	188.21	0.11	10.0
Upper vessel flow	kg/s	94.00	93.98	-0.02	10.0
		Steam Generator			
SG pressure - outlet	bar	64.5	64.50	0.0	0.1
Primary loss of pressure	bar	2.33	2.63	12.88	10.0
Feedwater temperature	K	491.15	491.15	0.0	0.5
Feedwater flow rate	kg/s	513.9	513.90	0.0	2.0
Steam mass flow	kg/s	513.9	512.34	-0.30	2.0
Recirculation mass flow	kg/s	1541.7	1541.3	-0.03	10.0
Liquid level	m	12.2	12.34	0.14 m	0.1 m
Thermal energy transferred	MW	945.5	944.99	-0.05	2.0
		Pressurizer			
Pressure	bar	-	158.41	-	0.1
Liquid Level	m	7.95	7.96	0.01 m	0.05 m
		Primary Circuit			
Hot-Leg Pressure	bar	158.0	158.11	0.07	0.1
Hot-Leg Temperature	K	599.25	600.72	0.25	0.5
Cold-Leg Temperature	K	564.45	566.29	0.33	0.5
Circuit mass flow	kg/s	4700.0	4699.70	-0.01	2.0
Total Pressure Loss	bar	6.5	6.37	-2.00	10.0

Table 4. Availability of ECCS components - LBLOCA

ECCS components	Injection							
Circuit	10		20		30		40	
Injection local (leg)	Hot	Cold	Hot	Cold	Hot	Cold	Hot	Cold
Safety injection pump	1	-	Break [a]	-	1	-	1	-
Accumulators	1	1	Break	SF [b]	R [c]	1	1	1
Residual heat removal pump	1		Break	SF	1		1	

a. Injected coolant lost via the break.
b. Single failure of isolation valve.
c. Repair case.

Table 5. Containment nodalization of Angra 2 - correspondence between the code components and the hydraulic zones

Hydraulic Region		Corresponding Component
	RPSUMP1	Sump[1]
	RPSUMP2	Sump
	R1	Containment
Containment	R2	Containment
	RDOME	Containment
	ANNUL	Annulus
	CONC	Secondary containment

Containment Nodalization

The COCOSYS V2.4 code version (Moody, 1965), was used for containment pressure calculation. This code can perform a complete containment simulation in case of base design accidents and even several accidents for Light Water Reactors (LWR). Four tables (evolution of mass flow and enthalpy of the phases - liquid and steam - and for each side of the break) make up the mass and energy additions from the primary depressurizing in case of a LBLOCA. With the pressure and temperature results obtained from the containment simulation it would also be possible to calculate with more accuracy the Peak of Cladding Temperature (PCT) values, the fuel temperature and the blowdown, refill and reflood periods.

The FSAR/A2 indicates this methodology for design base accidents study and uses S-RELAP5 and COCO codes, to simulate the entire Angra 2 plant and the containment conditions, respectively. Fig. 3 presents the simplified Angra 2 containment model for the LBLOCA simulation proposed on this paper with the COCOSYS code. Table 5 defines the zones on Fig. 3. During the COCOSYS nodalization development, it was observed that, in the case of a LBLOCA, the heat exchange structures details almost did not interfered on the containment pressure and temperature peak values, that occurred on the first seconds of the accidents considered. Therefore, we opted for a more simplified nodalization, since the containment pressure and temperature analyze considering iteration between the codes is the objective of this study.

RESULTS

FIG. 4, 5 and 6 show LBLOCA-HL, LBLOCA-crossover and LBLOCA-CL containment pressure distributions of accident first 250 seconds for the three cases described in the TAB. 2, respectively.

These COCOSYS simulations are compared with the RFAS/A2. The Base, Low, and High Cases distributions are close and surely below the design pressure value (6.3bar). The containment pressure peak of LBLOCA-HL, among all the cases analyzed in this study, was the largest found.

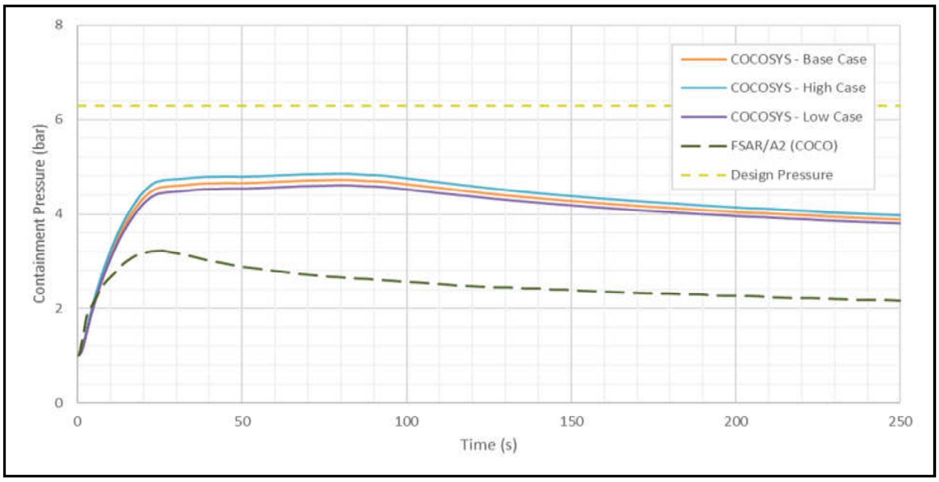

Figure 4. Containment pressure temporal distribution (LBLOCA-HL)

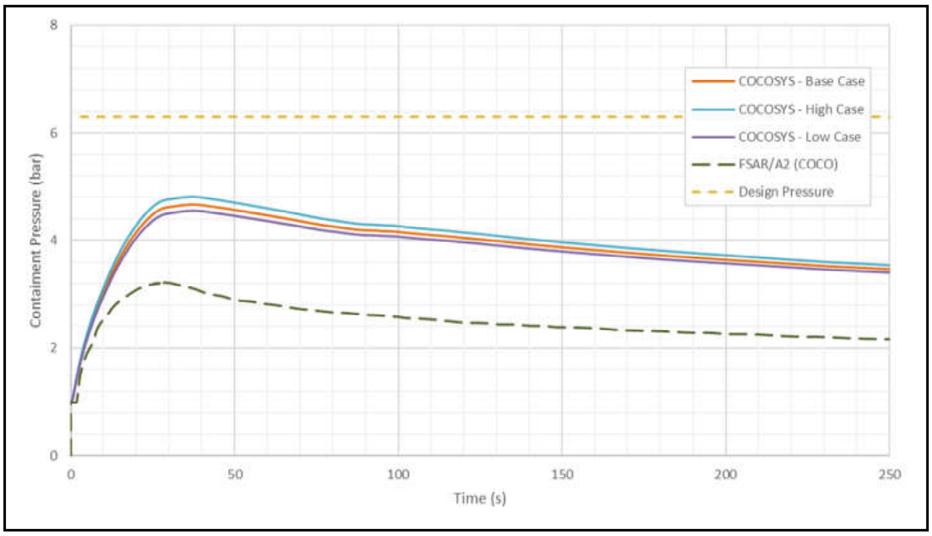

Figure 5. Containment pressure temporal distribution (LBLOCA-crossover)

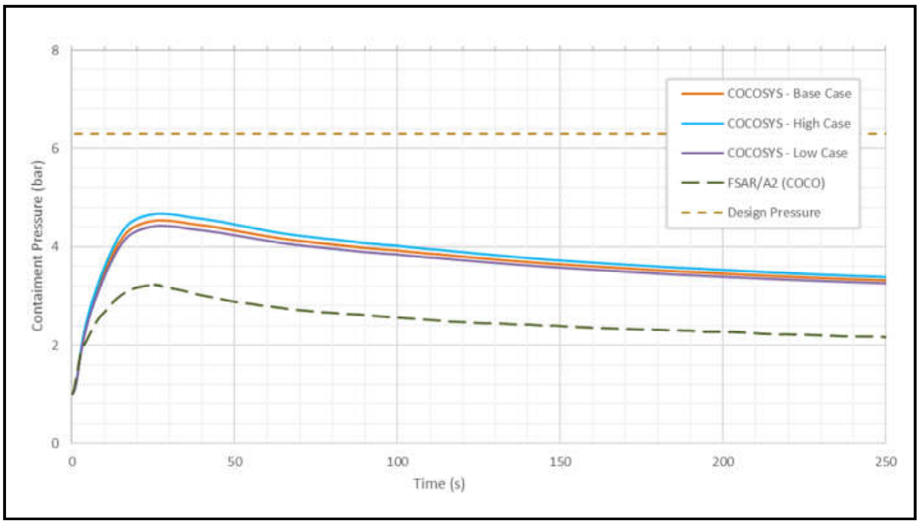

Figure 6. Containment pressure temporal distribution (LBLOCA-CL)

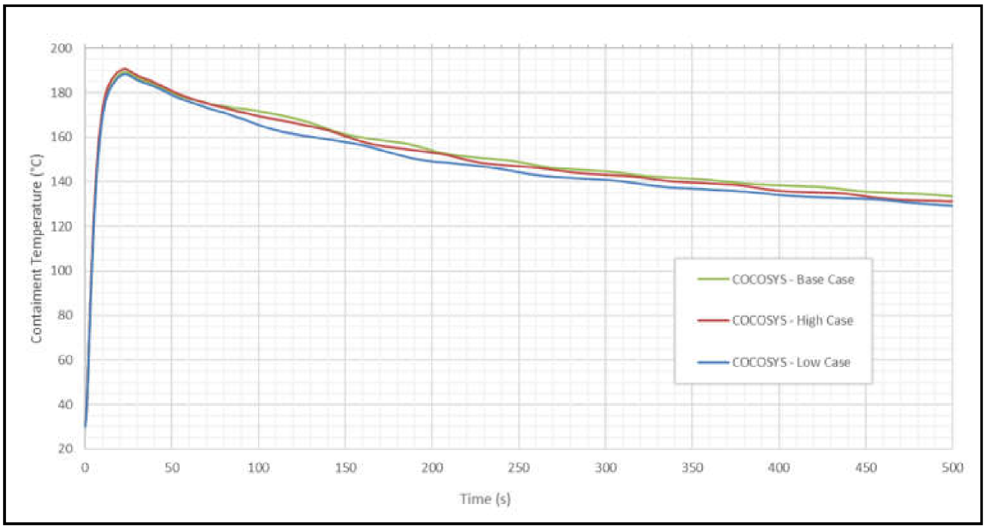

Figure 7. Containment temperature temporal distribution (LBLOCA-HL)

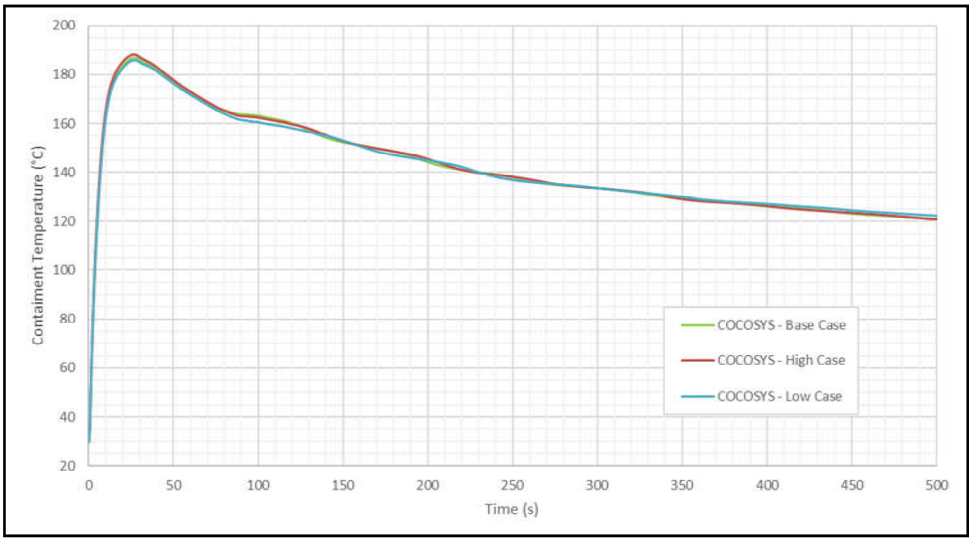

Figure 8. Containment temperature temporal distribution (LBLOCA-crossover)

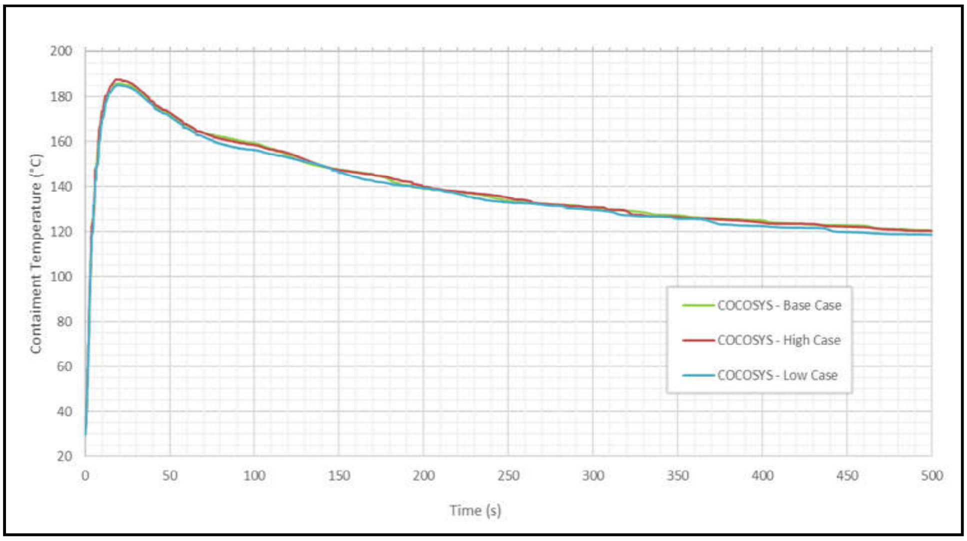

Figure 9. Containment temperature temporal distribution (LBLOCA-CL)

This occurs because the containment energy released is larger, when compared with the other two breaks. The pressure increases quickly for the first 26s of the accident, reaching the value of 4.71bar (Base Case). However, unlike the other cases, the pressure continues to increase until the 78s, reaching a peak, calculated by COCOSYS, of 4.78bar, posterior to that defined by COCO. Still, it is surely below the design pressure value.

The containment pressure peak of LBLOCA-crossover Base Case is 4,67 bar at 36 seconds of the accident evolution. For the LBLOCA-CL Base Case, the pressure reaches the maximum value of 4.54 bar at 28 seconds. Fig. 7, 8 and 9 present the containment temperature during the LBLOCA-HL, LBLOCA-crossover and LBLOCA-CL accidents for the first 500 seconds, respectively. The temperature maximum value reaches 189,2 ºC at 24 seconds (LBLOCA-HL), 186,8 ºC at 26 seconds (LBLOCA-crossover) and 185,8 ºC at 18 seconds (LBLOCA-CL).

Conclusions

The contribution of this work is a best estimated calculation of Angra 2 containment's pressure and temperature in the three LBLOCA cases analyzed. The Angra 2 containment pressure results obtained with COCOSYS code were satisfactory, since they were close to the FSAR/A2 results. The containment pressure values obtained with the GRS code are below the design pressure defined by the FSAR/A2 and the greatest containment pressure peak occurs on the hot leg rupture, due to the greater release of energy on time. In all cases, the pressure peaks were higher than those presented in FSAR/A2, because the accidents were simulated with a plant modeling and boundary conditions more conservative than those defined by FSAR/A2, which influenced the results of addition of mass and energy released to the containment.

Several phenomena can threaten the integrity of a nuclear reactor containment. Some significant accidents indicate the containment importance to retain radionuclides emitted, avoiding deleterious effects on the environment and population. Therefore, understand and predict such phenomena and avoid or minimize their consequences in the various projects of existing plants and future projects become necessary. The approach adopted on this paper corroborates the importance of using a more realistic methodology for the new PWR nuclear power plants evaluation, since computational tools and more realistic assumptions were adopted. Studies such this allow lower costs projections of new plants maintenance and operation.

Acknowledgments

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