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# CONTAINMENT PRESSURE ANALYSIS METHODOLOGY DURING A LBLOCA WITH ITERATION BETWEEN RELAP5 AND COCOSYS

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## ABSTRACT

The pressure conditions inside the containment in the case of a Large Break Loss of Coolant Accident (LBLOCA) are more severe in the case of hot leg rupture, due to the large amount of mass and energy that is thrown from the break that lies just after the pressure vessel. This work presents a methodology of pressure analysis within the containment of a Brazilian PWR, Angra 2, with an iterative process between the code that simulates guillotine rupture - RELAP5 - and the COCOSYS code, which analyzes the containment pressure from the accident conditions. The results show that the iterative process between the codes allows the convergence of pressure data to a more realistic approach.

## 1. INTRODUCTION

The International Atomic Energy Agency 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.” [1].

One of those accidents is the Loss of Coolant Accident (LOCA), defined as an accident that results in the loss of refrigerant that goes beyond the restoration capacity of the volumetric refrigeration control system [2]. One of the worst cases of this accident for the containment integrity is the LBLOCA in the Hot-Leg piping (LBLOCA-HL) - the maximum theoretical break area (2A, also known as double-ended guillotine break) of the primary between the Reactor Pressure Vessel (RPV) and Steam Generator (SG) - because the coolant mass and energy emission into the containment. After the blowdown, there is still release of mass produced by the Emergency Core Cooling System (ECCS).

It’s a requirement in the design of a nuclear plant that the containment building supports the pressures and temperatures resulted from this type of event [2]. 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 sufficient safety margin[1] for the entire sequence of events within the limits established by the regulatory body [1]. 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 Regulations (CFR) 50.46 could 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 [3].

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 [4], which follows Appendix K in the case of a LOCA, except that Best Estimate computational codes are used instead of conservative codes.

## 2. METHODOLOGY

The methodology used for the containment pressure analysis is presented in the flowchart below (Fig. 1).

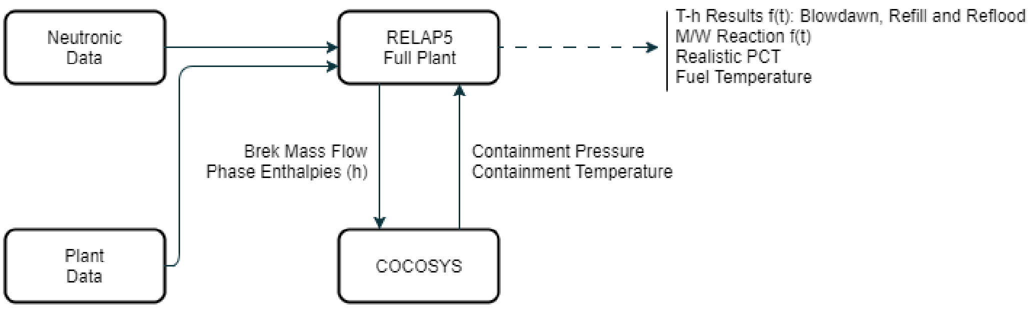


**Figure 1: Methodology used in the analysis of the containment with RELAP5 and COCOSYS codes.**

The COCOSYS V2.4 code was used to analyze the conditions in the containment of the Angra 2 reactor during a LBLOCA-HL. As boundary conditions, there was used the results of a simulation of this same accident, calculated by the RELAP5/MOD3.2Gamma. 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 considered the containment condition. Although not

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

analyzed in this work, the results of the core conditions would possibly improve when considering the iterative methodology between RELAP5 and COCOSYS codes.

### 2.1. 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 with a concrete dome, with diameter of 60m, thickness of 60cm and height of 60m [6].

The geometric and operational conditions of the Angra 2 containment considered, according to its Final Safety Analysis Report (FSAR/A2) [6], 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 | $7.1\text{x}10^4$ |
| Steel containment thickness | mm | 30.0 |
| Design manometric pressure | bar | 5,3 |
| Steel Containment Surface | m² | $7.66 \text{ x } 10^3$ |

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 [6].

### 2.2 Accident Description

The accident considered is the rupture in the hot leg of the primary circuit, between the outlet of the pressure vessel and the input of the steam generator circuit 20 (LBLOCA-HL). This accident is described in item 15.6.4.2.3.1.3[4] of the accident analysis chapter of FSAR/A2.

To obtain the containment pressure in the event, there was considered as initial condition of the LBLOCA-HL simulation the results of the Technical Report [7], which uses the basic input and nodalization developed by The Working Group of CNEN [8] for the simulation of such accident.

[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 *Double-Ended Hot-Leg Break.*

A nodalization was done for all four coolant loops of the primary circuits, but only one circuit (20), which contains the pressurizer, is presented in Fig. 2. That is the loop which the rupture was considered in this work, since its represent the worst scenario in LBLOCA-HL, due to the faster drainage of the surge line and the pressurizer.

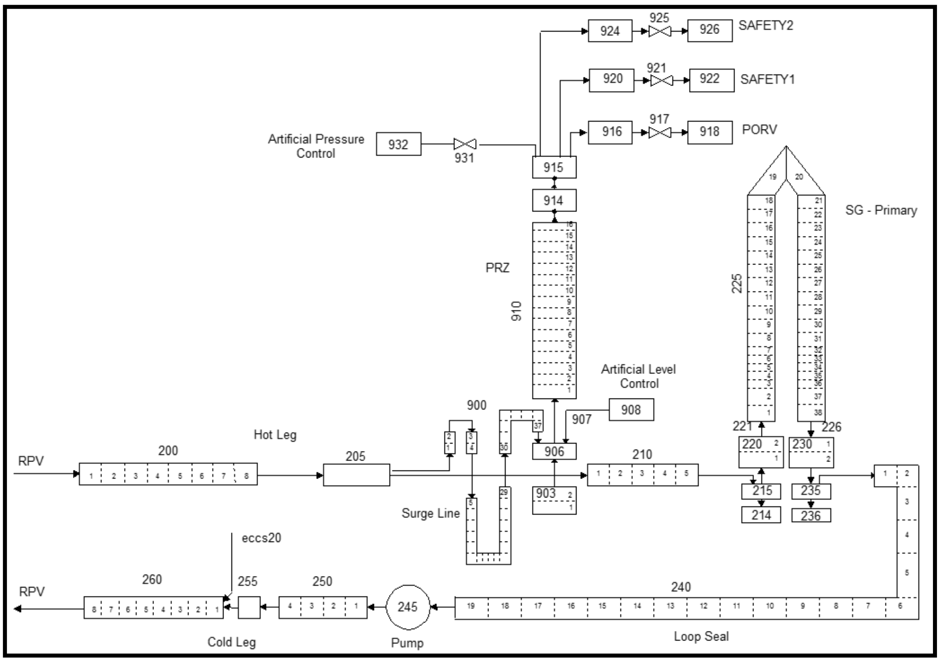


**Figura 2: Angra 2 loop 20 nodalization.**

Fig. 3 shows the location of the rupture (represented by the closure of valve V-953 and opening of valves V-951 and V-952) in a case of LBLOCA-HL for the proposed nodalization.

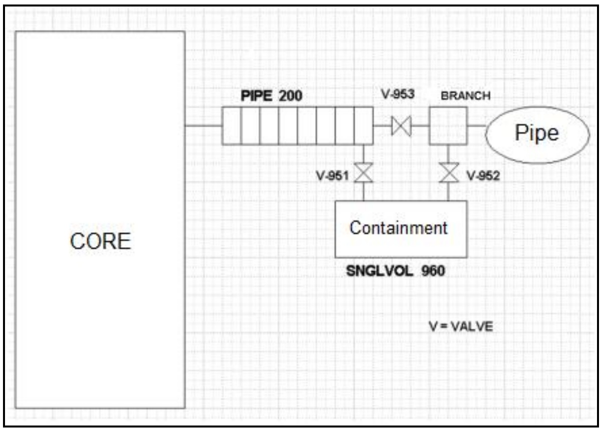


**Figura 3: Angra 2 primary circuit hotleg tubing break location ((LBLOCA-PQ).**

The initial and boundary conditions adopted in this simulation follow those specified in Table 2. Conservative approaches were chosen by assuming restrictive availability of the

ECCS with repairs and single failure affecting important components, as listed in Table 3, corresponding to FSAR/A2 Table 15.6.4.2-9 [6].

**Table 2: Initial conditions of the Angra 2**

| Parameter | Unit | Nominal [RFAS/A2] | Relap5/ Mod 3.2gama | Error (%) [5] | |
|---|---|---|---|---|---|
| | | | | 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 |

[5] D'Auria, F., Galassi, G. M., Belsito, S., Gatta, P., Ingegneri, M., *UMAE Application: Contribution to the OECD/CSNI UMS Vol. 2, Uncertainty Methods Study for Advanced Best Estimate Thermal Hydraulic Code Applications.* Vol. 2, NEA/CSNI/R(97)35, França, pp.2.I-2.114, 1998.

**Table 3: Availability of ECCS components, LBLOCA-HL**

| 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.

The accident was simulated with the code RELAP5/MOD3.2Gamma [9]. This code can simulate a LOCA by small, medium or large rupture. In addition, it can simulate transients of loss of electrical power, loss of feed water, loss of flow, among others. The analysis of thermohydraulic behavior during one of these accidents or transients applies both to the primary and secondary circuits of a nuclear plant.

This is a Best Estimated version of the RELAP5 code. One of the contributing factors is their discharge rate model, which allows to adopt the Henry Fauske model [10]. Studies [11] show that this model was less conservative than the Ransom-Trapp model [12] and Moody model [13] (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 [6].

### 2.3 Containment Nodalization

About the calculation of the conditions of pressure in the containment, it was used the COCOSYS V2.4 code version [13]. 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-HL.

The results of pressure and temperature obtained from the containment simulation were used on an iterative process for a new simulation of the LBLOCA-HL of the whole plant to obtain more realistic values of the pressure peaks in the containment during this event. With these results, it would also be possible to calculate with more accuracy the values of the Peak of Cladding Temperature (PCT), the temperature of the fuel and the blowdown, refill and reflood periods. This methodology is indicated in FSAR/A2 for the study of design base accidents, in which the S-RELAP5 code is used to simulate the entire Angra 2 plant and the COCO code to simulate the containment of this plant.

Fig. 4 presents the simplified Angra 2 containment model for the LBLOCA-HL simulation proposed in this work with the COCOSYS code. Table 4 defines the zones on Fig. 2.

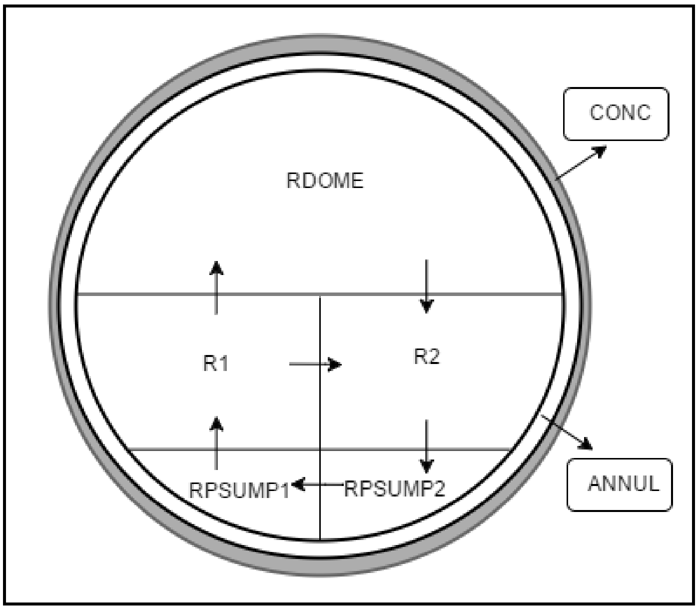


**Figura 4: Containment nodalization of Angra 2**

**Table 4: Containment nodalization of Angra 2 - correspondence between the components of the code and the hydraulic zones**

| Hydraulic Region | | Corresponding Component |
|---|---|---|
| Containment | RPSUMP1 | Sump[6] |
| | RPSUMP2 | Sump |
| | R1 | Containment |
| | R2 | Containment |
| | RDOME | Containment |
| | ANNUL | Annulus |
| | CONC | Secondary containment |

In the development of the COCOSYS nodalization, it was observed that, for the case of a LBLOCA-HL, the detail of the heat exchange structures little interfered in the calculation of the peaks of containment pressure and temperature, that occur in the first seconds of the accidents considered. Therefore, we opted for a more simplified nodalization, since the objective of this work is to observe the results of pressure in the containment when there is iteration between the codes.

[6] Sump is part of the Emergency Core Cooling System (ECCS). In the case of a LOCA, the refrigerant flows to the sump, thus serving as recirculating water source, waste heat removal and emergency cooling of the core.

## 3. RESULTS

Fig. 5 shows the results of containment pressure of the accident first 250 s for three iterations between RELAP5 and COCOSYS. It also shows the containment pressure distribution indicated in the FSAR/A2 (COCO code) in case of a LBLOCA-HL and the limit of the containment design pressure (6.3bar).

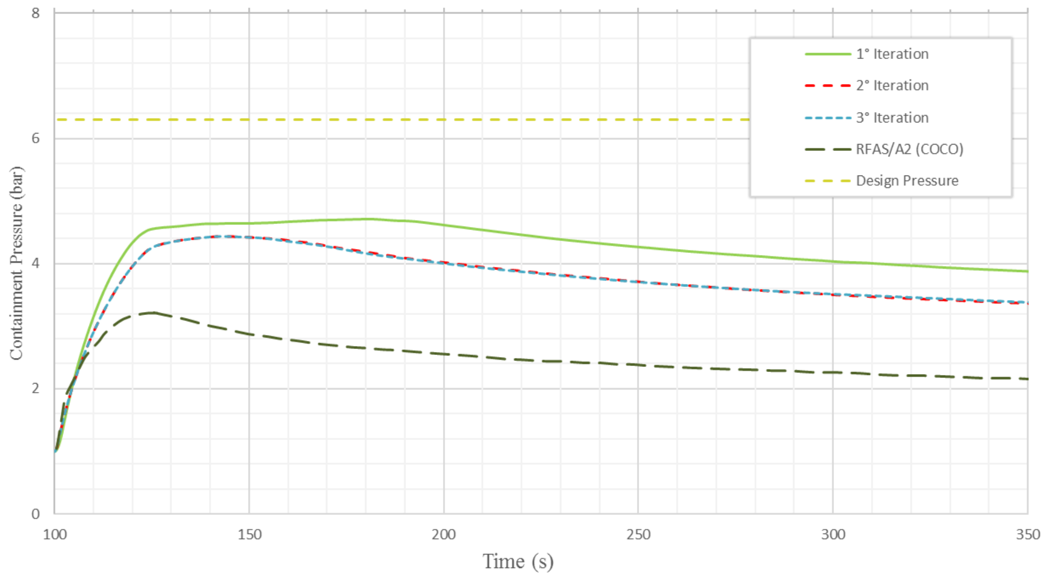


**Figura 5: Distribution of containment pressure during a LBLOCA-HL (iterative process).**

In the first iteration, the pressure increases fast to the first 26 s of accident, reaching the value of 4.71 bar, and continues increasing until the 78 s, reaching a peak calculated by the COCOSYS of 4.78 bar. It's significantly later than the one defined by the COCO code. Still, it is surely below the design pressure value. However, on the second iterative process, the pressure distribution approaches the one indicated in FSAR/A2.

A third iteration was the last performed. This distribution converged with the pressure distribution of the previous iteration.

## 4. CONCLUSIONS

The contribution of this work is proposing a more realistic calculation of the Angra 2 containment pressure by the process of iteration between the thermohydraulic accident simulation code (RELAP5) and the code that calculates the conditions in the containment (COCOSYS). Despite a simplified nodalization, the containment pressure results for Angra 2 were satisfactory, since they are below of the containment design pressure and approach to the values and behavior of the one of FSAR/A2.

The iteration process was also satisfactory; besides it projects a more Best Estimated situation, it came closer to the FSAR/A2 values.

The approach adopted in this paper corroborates the importance of using a more realistic methodology for the evaluation of new PWR nuclear power plants, since computational tools and more realistic assumptions were adopted. Studies of this type allow lower costs projections of new plants maintenance and operation.

## ACKNOWLEDGMENTS

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