Associação Brasileira de Energia Nuclear – ABEN
2019 International Nuclear Atlantic Conference - INAC 2019
Santos, SP, Brazil, October 21-25, 2019

Agência Brasileira do ISBN
ISBN 978-85-99141-08-3

SMALL BREAK LOSS OF COOLANT ACCIDENT OF 200 cm² IN COLD LEG OF PRIMARY LOOP OF ANGRA 2 NUCLEAR POWER REACTOR EVALUATION

Eduardo Madeira Borges and Gaianê Sabundjian

Instituto de Pesquisas Energéticas e Nucleares (IPEN / CNEN - SP)
Av. Professor Lineu Prestes 2242
05508-000 São Paulo, SP
borges.em@hotmail.com
gdjian@ipen.br

ABSTRACT

The aim of this paper is evaluated the consequences to ANGRA 2 nuclear power reactor and to identify the flow regimes, the heat transfer modes, and the correlations used by RELAP5/MOD3.2.gama code in ANGRA 2 during the Small-Break Loss-of-Coolant Accident (SBLOCA) with a 200cm^2 of rupture area in the cold leg of primary loop. The Chapter 15 of the Final Safety Analysis Report of ANGRA 2 (FSAR-A2) reports this specific kind of accident. The results from this work demonstrated the several flow regimes and heat transfer modes that can be present in the core of ANGRA 2 during the postulated accident. The results obtained for ANGRA 2 nuclear reactor core during the postulated accident were satisfactory when compared with the FSAR-A2. Additionally, the results showed the correct actuation of the ECCS guaranteeing the integrity of the reactor core.

1. INTRODUCTION

The evaluated accident consists of the partial break of the cold leg of the ANGRA 2 nuclear power plant. The rupture is the 200 cm^2 and the efficiency of the Emergency Core Coolant System (ECCS) is verified for this accident.

The aim of this paper is to identify the flow regimes, the heat transfer modes, and the correlations used in the RELAP5/MOD3.2.gama [1] code in ANGRA 2 during the Small-Break Loss-of-Coolant Accident (SBLOCA) with a 200cm^2 of rupture area in the cold leg of primary loop how described in detail in the Chapter 15 of the Final Safety Analysis Report of ANGRA 2 (FSAR-A2) [2].

Results presented in this paper showed the correct actuation of the ECCS guaranteeing the integrity of the ANGRA 2 reactor core.

2. ANGRA 2 NUCLEAR POWER REACTOR

ANGRA 2 is the second Brazilian nuclear power plant located at the Central Nuclear Almirante Álvaro Alberto (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 (now 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.

ANGRA 2 has four pumps to control of water flow, four loops with two ECCS (Emergency Core Cooling System) for each loop (one Hot and one Cold ECCS). Figure 1 shows the arrangement of the components of ANGRA 2 nuclear power plant [3].

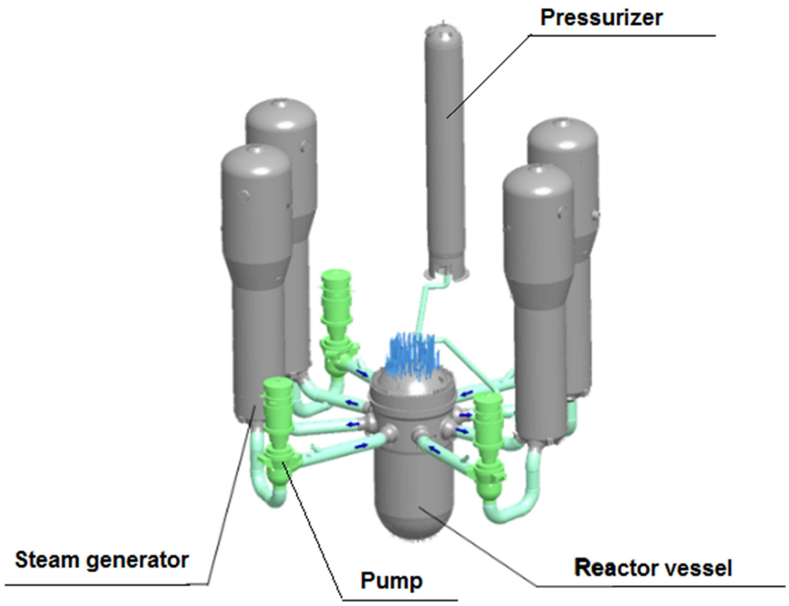

Figure 1: Arrangement of the ANGRA 2 nuclear power plant components.

3. RELAP5 CODE

The RELAP5 code was developed by the Idaho National Laboratory. This code was originally designed for the analysis of thermal hydraulic transients in Pressurized Water Reactors (PWR). The RELAP5 can model the primary and secondary cooling systems of experimental facilities and of Nuclear Reactors with geometric details. The program uses the non-homogeneous non-equilibrium two-fluid model, and considers the conservation equations of mass, momentum and energy for the liquid and gas phases. One-dimensional model is used to treat the fluid flow and the heat conduction in the structures; however, in some special cases such as the cross flow in the reactor core and the rewetting region in flooding model, the two-dimensional model is used [1].

The RELAP5 code uses and is capable to identify fifteen different flow regimes, which are presented in Table 1. Each one associated to an integer number. Those numbers are obtained from RELAP5 code output file to specify the fluid behavior for each control volume during the accident simulation [1].

Tables 2 and 3 show the mode numbers and the wall convection heat transfer used in RELAP5 code, respectively [1]. They were accessed during the execution of the program to this case, and the results are presented in the next item of this paper.

Table 1.: Flow regime number

Flow Regime	Number
High mixing bubbly	1
High mixing bubbly/mist transition	2
High mixing mist	3
Bubbly	4
Slug	5
Annular mist	6
Mist pre-CHF	7
Inverted annular	8
Inverted slug	9
Mist	10
Mist post-CHF	11
Horizontal stratified	12
Vertical stratified	13
Level tracking	14
Jet junction	15

Table 2.: Correspondent numbers of RELAP5 flow modes

Number	Mode
0	Convection to noncondensable-water mixture
1	Single-phase liquid convection at supercritical pressure
2	Single-phase liquid convection, subcooled wall, low void fractions
3	Subcooled nucleate boiling
4	Saturated nucleate boiling
5	Subcooled transition boiling
6	Saturated transition boiling
7	Saturated film boiling
8	Saturated film boiling
9	Single-phase vapor convection or supercritical pressure with the void fraction greater than zero
10	Condensation when the void is less than one
11	Condensation when the void equals one

Table 3.: Wall convection heat transfer mode numbers

Mode Number	Heat Transfer Phenomena	Correlations References
0	Noncondensable-steam- water	[4, 5, 6, 7, 8, 9].
1	Supercritical or single-phase liquid	Same as mode 0.
2	Single-phase liquid or subcooled wall with voidg < 0.1	Same as mode 0.
3	Subcooled nucleate boiling	[10].
4	Saturated nucleate boiling	Same as mode 3.
5	Subcooled transition boiling	[11].
6	Saturated transition boiling	Same as mode 5.
7	Subcooled film boiling	[4, 5, 6, 7, 8, 9, 12, 13].
8	Saturated film boiling	Same as mode 7.
9	Supercritical two-phase or single-phase gas	Same as mode 0.
10	Film wise condensation	[14, 15, 16].
11	Condensation in steam	Same as mode 10.
3 & 4	Horizontal bundles with nucleated boiling	[4, 17, 18].

4. ACCIDENT SIMULATION AND RESULTS

The accident started after 100 seconds of the steady state simulation time, when the valve 951 was opened. Valve 951 is connected to the branch 255 (primary cold leg), which is connected to the volume 960 (containment). The area of the valve opening is 200 cm^2. This is the size of the simple rupture considered in this case.

The input file was based in the work performed by the Technical Cooperation among *Instituto de Pesquisas Energética e Nucleares* (IPEN), *Centro de Desenvolvimento Tecnológico Nuclear* (CDTN), and *Comissão Nacional de Energia Nuclear* (CNEN) [19, 20].

Figure 2 shows the nodalization of the ANGRA 2 reactor core. Although, the cooling primary loop was modeled in the simulation using RELAP5, it is not presented in the figure 2. The boundary conditions used were taken from FSAR-A2 [2].

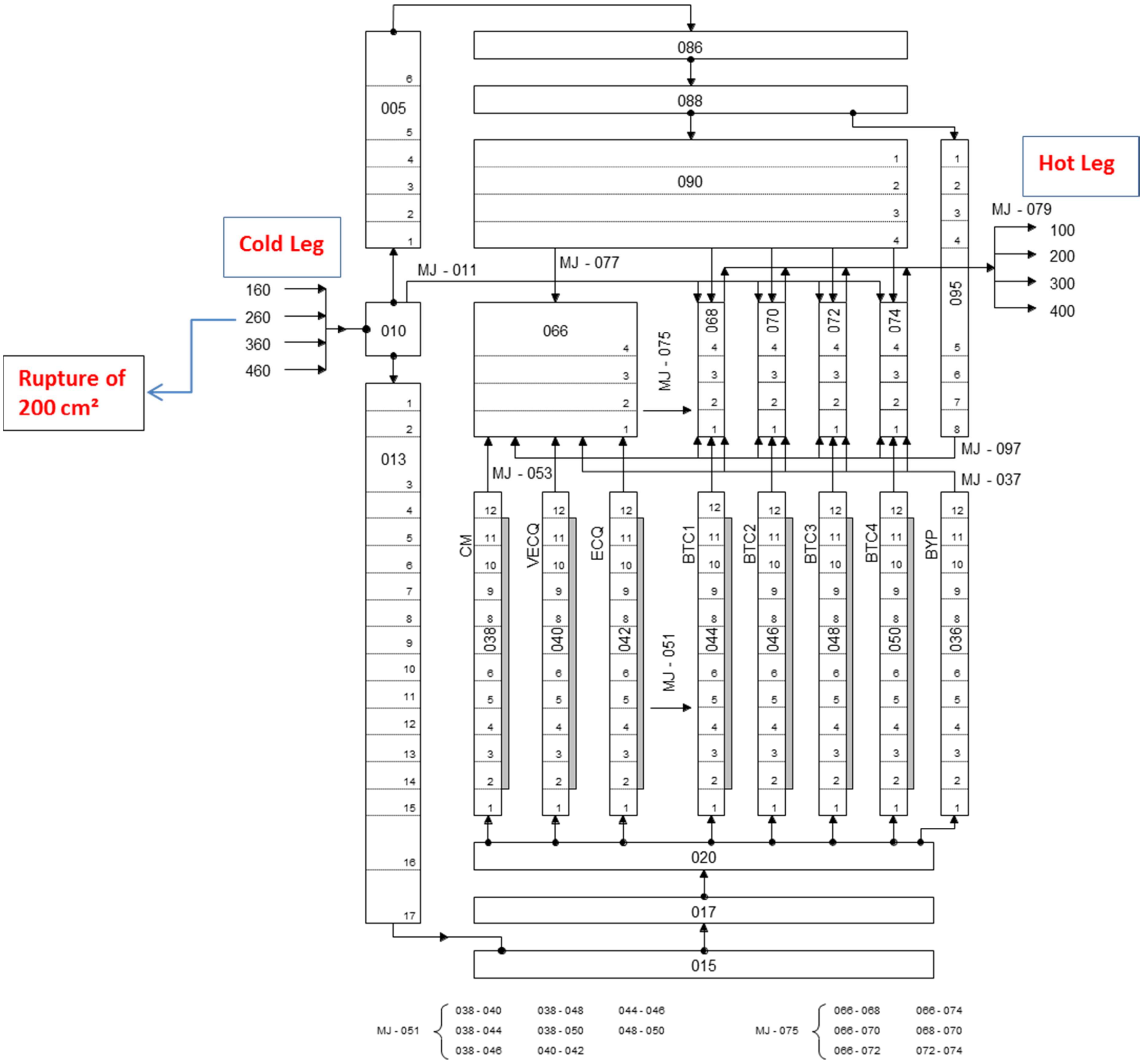

Figure 2: ANGRA 2 nuclear reactor core nodalization to RELAP5 code.

The main boundary conditions used in this simulation were obtained from the FSAR-A2 [2] and are presented as the following:

- reactor power - 106% nominal power;
- reactor trip from Reactor Coolant System (RCS) pressure < 132 bar;
- 100 k/h secondary-side cooldown (P_{RCS} < 132 bar and containment pressure > 1.03 bar);
- ECC criteria met (P_{RCS} < 110 bar and containment pressure > 1.03 bar).

For each postulated LOCA, the ECCS performance is different. The Chapter 15 of the Final Safety Analysis Report of ANGRA 2 (FSAR-A2) reports the ECCS actuation for each accident [2]. In this case failure and repair criteria for the ECCS components were adopted as specified to this event in the FSAR-A2 in order to verify the system operation, preserving the integrity of the reactor core and to guarantee its cooling, as presented in Table 4.

Table 4.: Injection by the ECCS for SBLOCA

ECCS Components	Injection							
	Loop 10		Loop 20		Loop 30		Loop 40	
	hot	cold	hot	cold	hot	cold	hot	cold
Safety Injection Pumps	1	_	1	_	SF	_	RC	_
Accumulators	1	1	1	1	1	1	1	1
Residual Heat Removal Pumps	1		1		SF		RC	

SF: Single failure of diesel engine, RC: Diesel engine down for repairs.

SBLOCA accidents are characterized by a slow blow down in the primary circuit to values that the high pressure injection system is activated. The thermal-hydraulic processes inherent to the accident phenomenon, such as hot leg of ECCS vaporization and consequently core vaporization causing an inappropriate flow distribution in the reactor core, can lead to a reduction in the core liquid level, until the ECCS is capable to refill it. These are the principal reasons to identifying and understand the flow regimes and the heat transfer modes used by RELAP5 code in the nuclear core of ANGRA 2 during accident simulation.

Table 5 provides a summary of the analyzed accident, the temporal sequence of operation and evaluation of ANGRA 2 nuclear reactor ECCS performance. According to some results provided by RELAP5 and FSAR-A2 it was possible to observe the differences related to onset time of some phenomena.

Figures 3 to 12 show the results obtained from SBLOCA of ANGRA 2 analysis using RELAP5 code. Some of these results were compared with the results found in the FSAR-A2. Some results obtained using RELAP5 were similar to the results of the FSAR-A2 [2].

Figure 3 shows the pressures in the primary and secondary loops to RELAP5 and FSAR-A2. It is note that in RELAP5 code simulation the primary pressure decreases faster than FSAR-A2 one.

Table 5.: SBLOCA 200 cm² accident temporal sequence

EVENTS	TIME (s)	
	RELAP5	RFAS-A2
Break initiation	100	100
Reactor trip from RCS pressure (p_{RCS} < 132 bar): → turbine trip, loss of offsite power reactor coolant pump trip.	115	103.1
100 K/h secondary-side cooldown (p_{RCS} < 132 bar and p_{cont} > 1,03 bar)	115	103.1
ECCS criteria met (p_{RCS} < 110 bar and p_{cont} > 1,03 bar)	125	111
Safety injection pumps start (High pressure pump start)	155	141
Accumulator injection starts	930	750
Low pressure injection pumps start	980	1093
Hot rod PCT- Time - (°C)	515 - (759.5°C)	421 – (492.0°C)
Hot channel recovered	1260	1290
Cold-leg accumulators isolated (500 s after ECCS criteria signal)	625	611
The end of Simulation	1300	1300

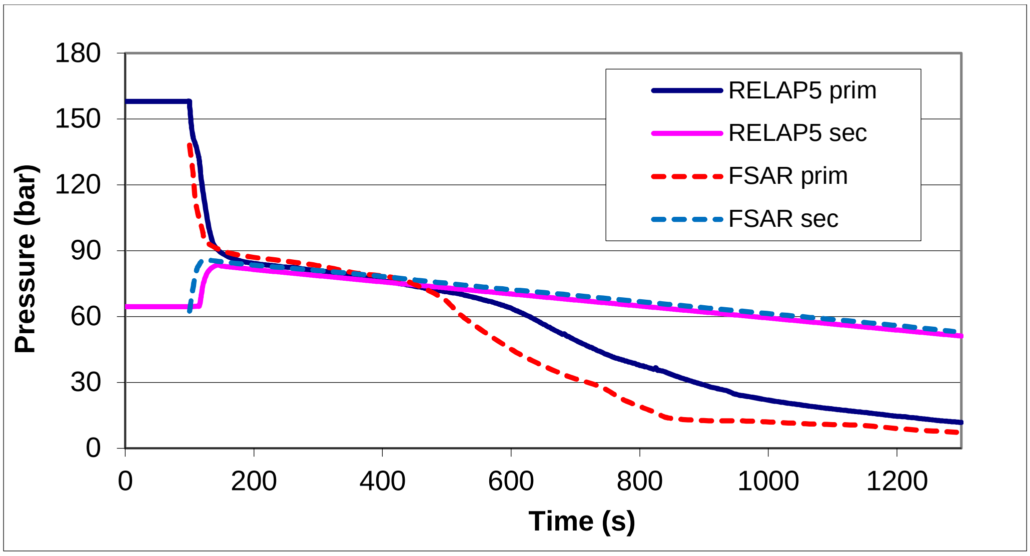

Figure 3: Pressure in the primary and secondary loops of ANGRA 2.

The ECCS system operates in function the primary pressure, therefor the ECCS to RELAP5 code simulation is fast than FSAR-A2 one. Figures 4 and 5 show the mass flow of ECCS lines to RELAP5 and FSAR-A2. Can be notes that mass flow rate of ECCS cold line to loop 30 and 40 is zero all time.

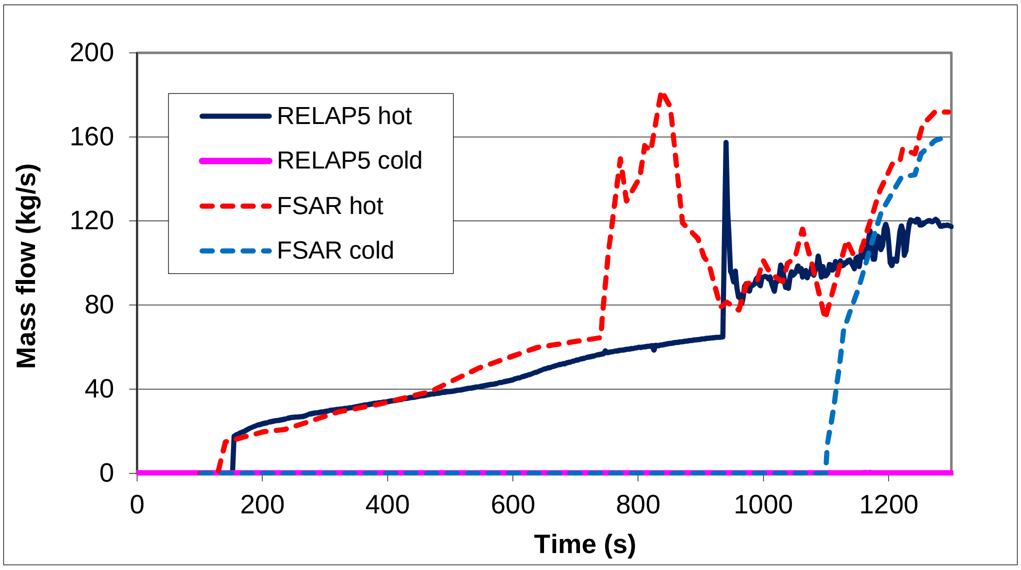

Figure 4: Mass flow in the lines of ECCS – Loops 10 and 20.

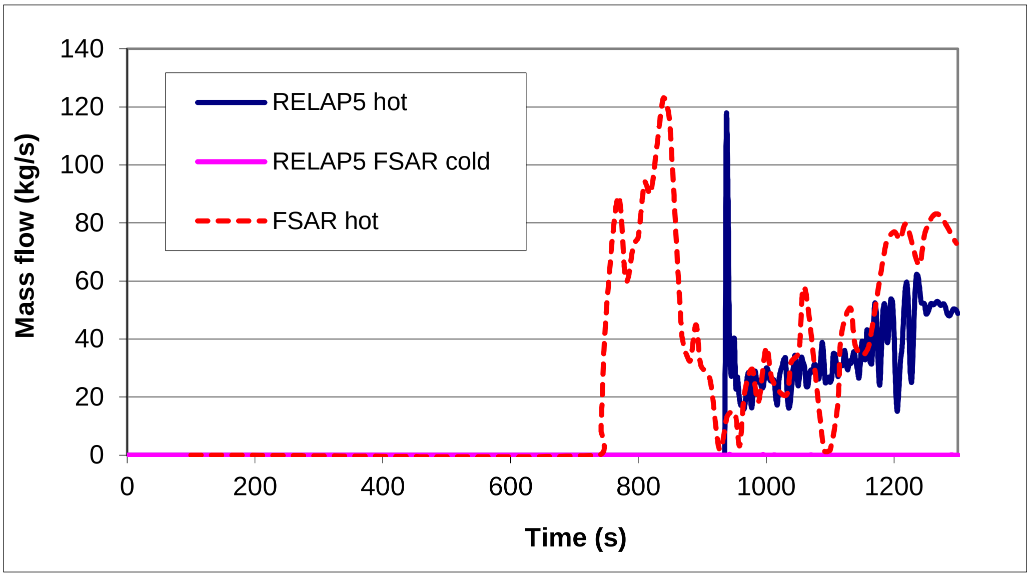

Figure 5: Mass flow in the lines of ECCS – Loops 30 and 40.

Figures 6 and 7 show the mass flow in the rupture and the primary loop coolant mass inventory, respectively. Note that the primary loop coolant mass inventory is the result of the sum of the mass flow of ECCS system minus the mass flow in the rupture.

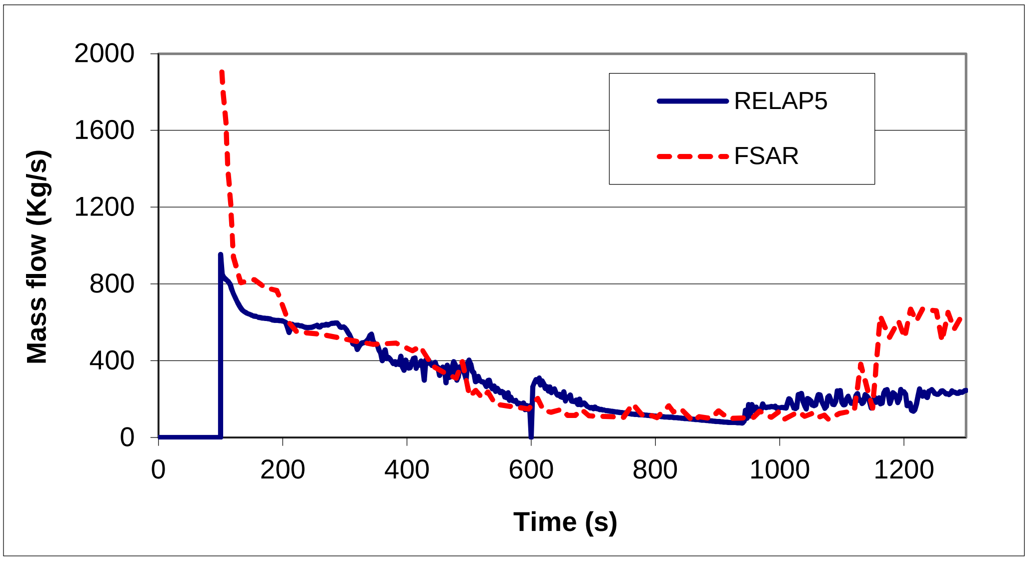

Figure 6: Mass flow in the break.

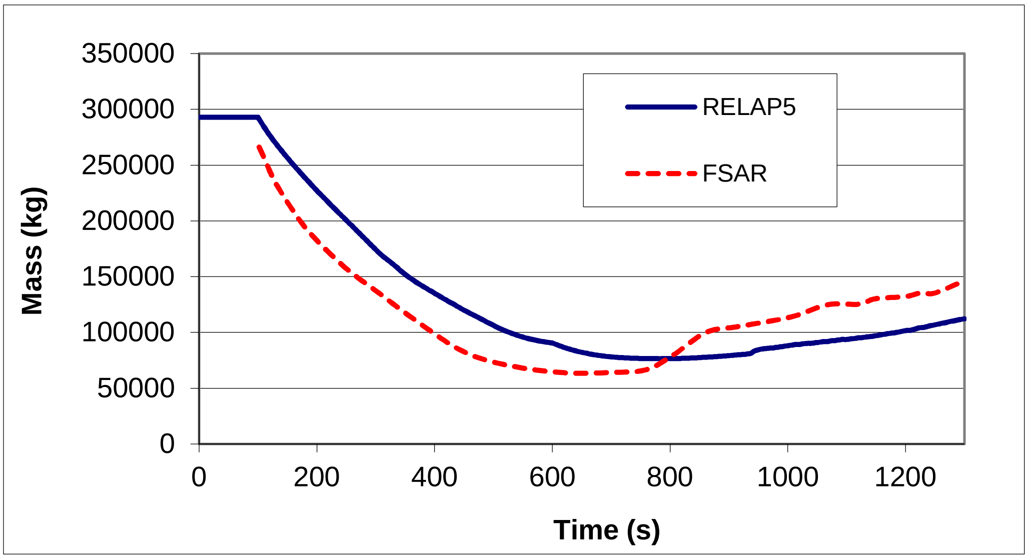

Figure 7: Primary coolant mass inventory.

Figure 8 shows the void fraction in the rupture. Between 630 and 930 seconds of simulation, only vapor in the rupture was observed to RELAP5 simulation. Note that after 630 seconds the void fraction to RELAP5 simulation is higher than FSAR-A2 one.

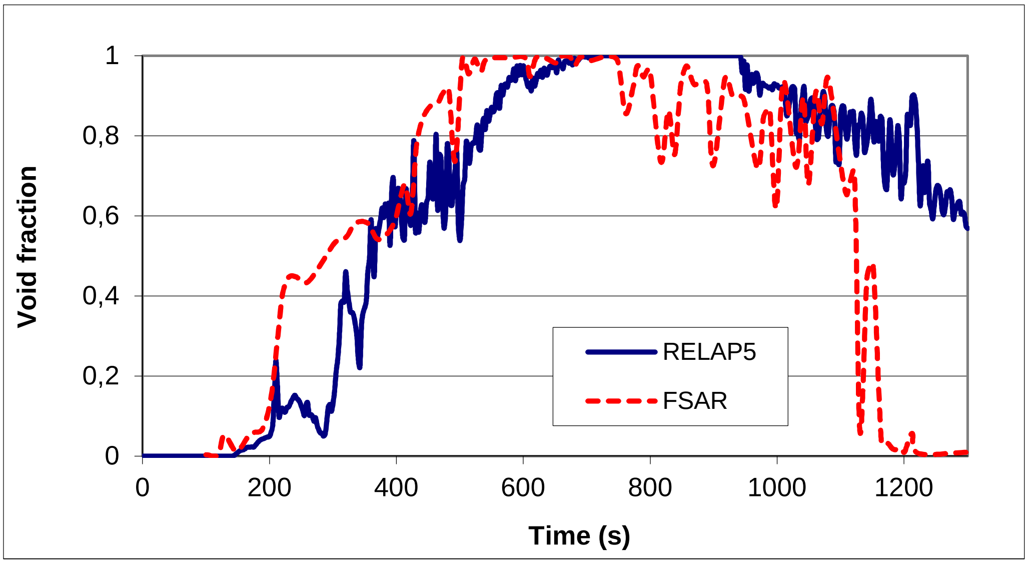

Figure 8: Void fraction in the break.

Figure 9 shows void fraction in the upper region of the hot channel of the core of ANGRA 2. Between 360 and 515 seconds, only vapor as observed to RELAP5 simulation. There aren´t FSAR-A2 core void fraction data.

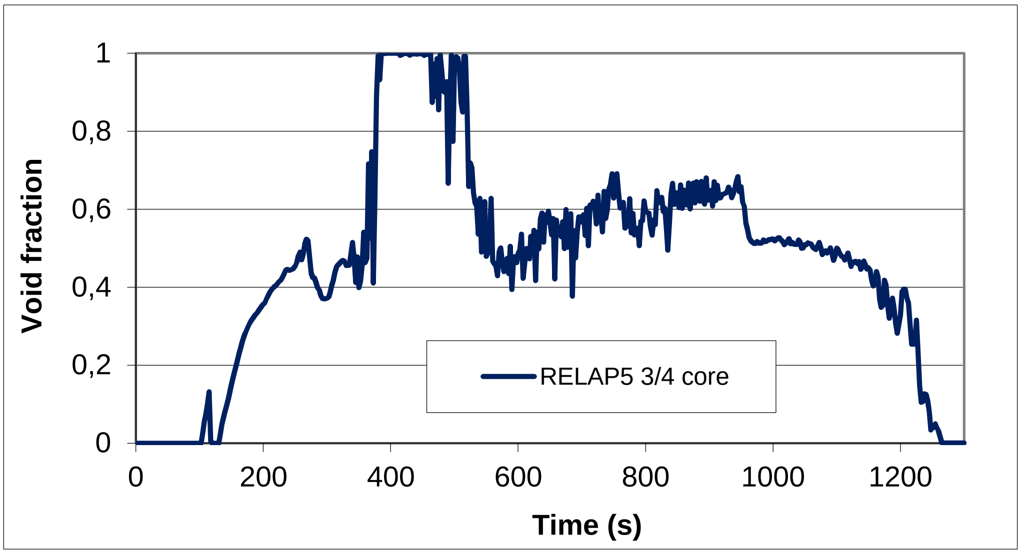

Figure 9: Void fraction to hot channel core of ANGRA 2.

Thermal hydraulic conditions, include local void fraction define which correlations are used by RELAP5 code. Figures 10 and 11 show the numbers of flow regimes and heat transfer modes on the upper region of the hot channel of the core, respectively, during the simulation using RELAP5 code. These variables and correlations used can be observed in the Tables 1 and 3.

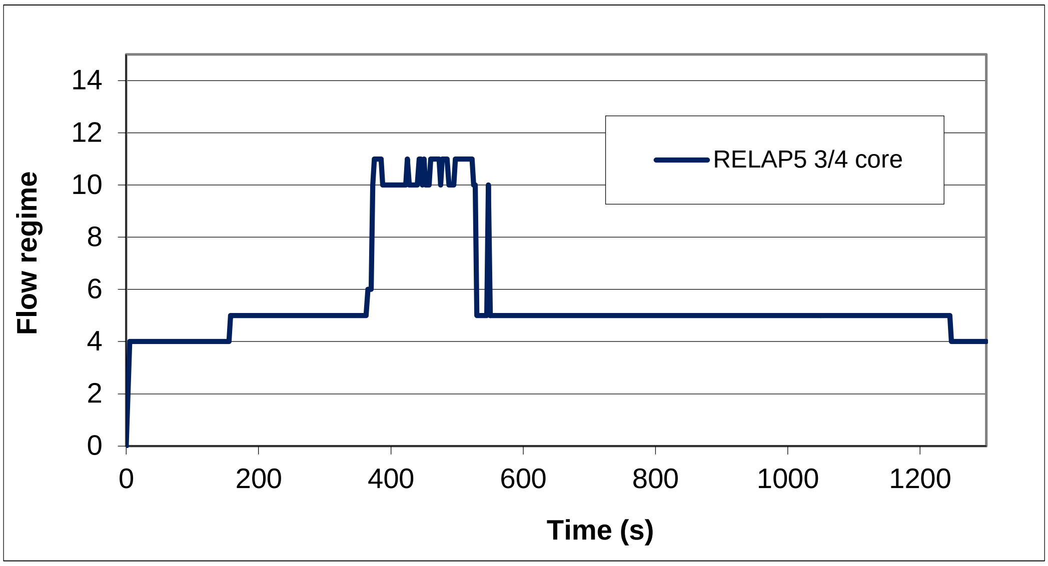

Figure 10: Flow regimes to hot channel core of ANGRA 2.

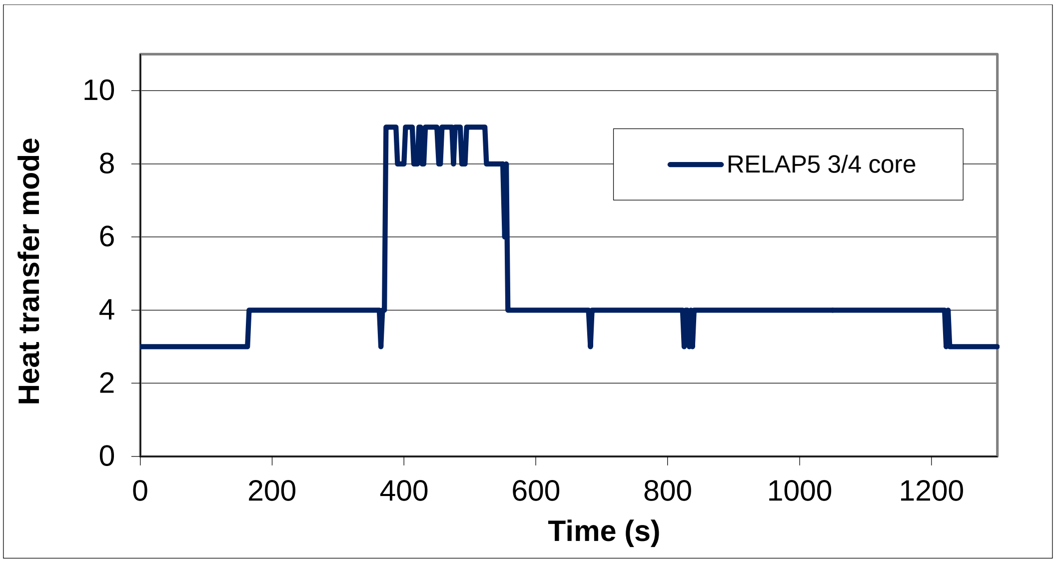

Figure 11: Heat transfer modes to hot rod cladding core of ANGRA 2.

Figure 12 shows three points of hot rod core cladding temperature of ANGRA 2 to RELAP5 simulation and FSAR-A2. These RELAP5 data are higher than FSAR-A2 one. But the most hot rod core cladding temperature is lower than 800 °C.

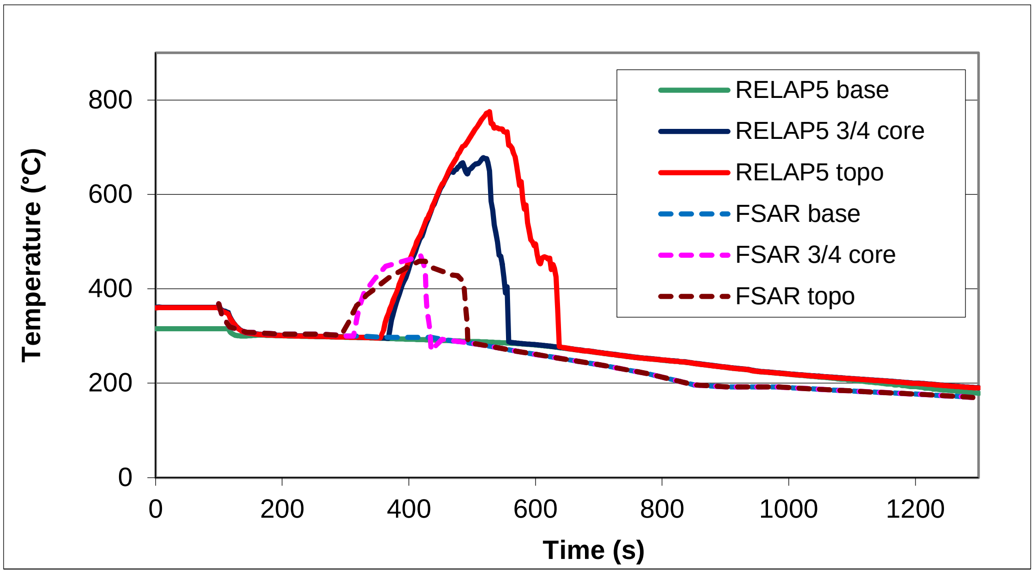

Figure 12: Hot rod cladding temperature of ANGRA 2 core.

5. CONCLUSIONS

In this work the flow regimes, the heat transfer modes, and the correlation used by RELAP5/MOD3.2.gama code, during the SBLOCA with 200cm^2 of rupture area in the cold leg of primary loop were identified.

The evaluation of the most important variables in this simulated accident with RELAP5 code, when compared to their FSAR-A2 data one, showed that the analysis of RELAP5 was more conservative than the FSAR-A2.

Results presented in this paper showed the correct actuation of the ECCS guaranteeing the integrity of ANGRA 2 reactor core.

REFERENCES

1. Idaho Lab. Scientech Inc., "RELAP5/MOD3 Code Manual Volume II: Appendix A Input Requirements, NUREG/CR-5535" – **Vol. II App A**, (1999).
2. Eletronuclear S. A., "Final Safety Analysis Report – Central Nuclear Almirante Álvaro Alberto – Unit 2", *Doc: MA/2-0809.2/060000 -Rev. 3,* (2000).
3. Mitsubishi Heavy Industries, "Robot Technologies of PWR for Nuclear Power Plant Maintenance", EJAM, Vol. 5, No 1, NT54, Available online at: http://www.jsm.or.jp/ejam/Vol.5No.1/NT/NT54/NT54.html, (2009).
4. ESDU Engineering Science Data Unit, 73031, *ESDU International Plc*, 27 Corshan Street, London, N1 6UA, (1973).
5. Kays, W.M. "Numerical Solution for Laminar Flow Heat Transfer in Circular Tubes", *Transactions, American Society of Mechanical Engineers, 77,* pp. 1265-1274, (1955).
6. Dittus, F.W. and Boelter, L.M.K. "Heat Transfer in Automobile Radiators of the Tubular Type", *Publications in Engineering, 2,* University of California, Berkeley, pp. 443-461, (1930).

7. Shah, M.M. *Heat Transfer and Fluid Flow Data Books,* Genium Publishing, Section 507.6 page 7, (1992).
8. Churchill, S.W. and Chu, H.H.S. “Correlating Equations for Laminar and Turbulent Free Convection from a Vertical Plate”, *International Journal of Heat and Mass Transfer, 18,* pp. 1323-1329, (1975).
9. McAdams, W.H. *Heat Transmission,* 3rd edition, New York: McGraw-Hill, (1954).
10. Chen, J.C. “A Correlation for Boiling Heat Transfer to Saturated Fluids in Convective Flow”, *Process Design and Development, 5*, pp. 322-327, (1966).
11. Chen, J.C.; Sundaram, R.K. and Ozkaynak, F.T. *A Phenomenological Correlation for Post-CHF Heat Transfer,* NUREG-0237, (1977).
12. Bromley, L.A. “Heat Transfer in Stable Film Boiling”, *Chemical Engineering Progress, 46*, pp. 221-227, (1950).
13. Sun, K.H.; Gonzales-Santalo, J.M. and Tien, C.L. “Calculations of Combined Radiation and Convection Heat Transfer in Rod Bundles under Emergency Cooling Conditions”, *Journal of Heat Transfer*, pp. 414-420, (1976).
14. Nusselt, W. “Die Oberflachenkondensation des Wasserdampfes”, *Ver. deutsch. Ing., 60,* (1916).
15. Shah, M.M. “A General Correlation for Heat Transfer during Film Condensation Inside Pipes”, *International Journal of Heat and Mass Transfer*, *22*, pp. 547-556, (1979).
16. Colburn, A.P. and Hougen, O.A. “Design of Cooler Condensers for Mixtures of Vapors with Noncondensing Gases”, *Industrial and Engineering Chemistry, 26*, pp. 1178-1182, (1934).
17. Forster, H.K. and Zuber, N. “Dynamics of Vapor Bubbles and Boiling Heat Transfer”, *AIChE Journal*, *1,* No. 4, pp. 531-535, (1955).
18. Polley, G.T.; Ralston, T. and Grant, I.D.R. “Forced Cross Flow Boiling in an Ideal In-line Tube Bundle”, *ASME 80-HT-46*, (1981).
19. D. A. Andrade, G. Sabundjian, Qualificação a nível transiente da nodalização a2nb03c: Acidente de SBLOCA de 380 cm2 da linha do sistema de resfriamento de emergência do núcleo (SREN), conectada à perna quente, P&D.CENT.CENT.005.00, RELT.002.R00, Instituto de Pesquisas Energéticas e Nucleares, São Paulo, (2001).
20. R. C. Borges, A. A. Madeira, L. C. M. Pereira, E. T. Palmieri, C. V. G. Azevedo, N. S. Lapa, G. Sabundjian e D. A. Andrade, “Simulação de Angra 2 com o código RELAP5/MOD3.2gamma”, *Sessão Técnica Especial*, *XIII Encontro Nacional de Física de Reatores e Termo-hidráulica*, Rio de Janeiro, RJ, Brasil, 11-16 de agosto, (2002).