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IDENTIFICATION OF FLOW REGIMES AND HEAT TRANSFER MODES IN ANGRA2 CORE DURING THE SIMULATION OF THE SMALL BREAK LOSS OF COOLANT ACCIDENT OF 250 cm^2 IN THE COLD LEG OF PRIMARY LOOP USING RELAP5 CODE

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ABSTRACT

The aim of this paper is 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 250cm^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 Angra2 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 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 250cm^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].

The accident consists of the partial break of the cold leg of the ANGRA 2 nuclear power plant. The rupture is the 250 cm^2 and the efficiency of the Emergency Core Coolant System (ECCS) is also verified for this accident.

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.

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

2. THE NODALIZATION OF ANGRA 2 USING THE RELAP5 CODE

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].

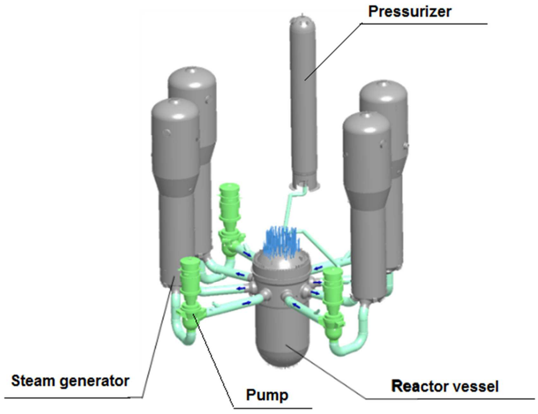

Figure 1: Arrangement of the ANGRA 2 nuclear power plant components.

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 [2] for each accident. 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 1.

Table 1: 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

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. The boundary conditions used were taken from FSAR-A2. The accident started after 100 seconds of the steady state simulation time.

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) [4].

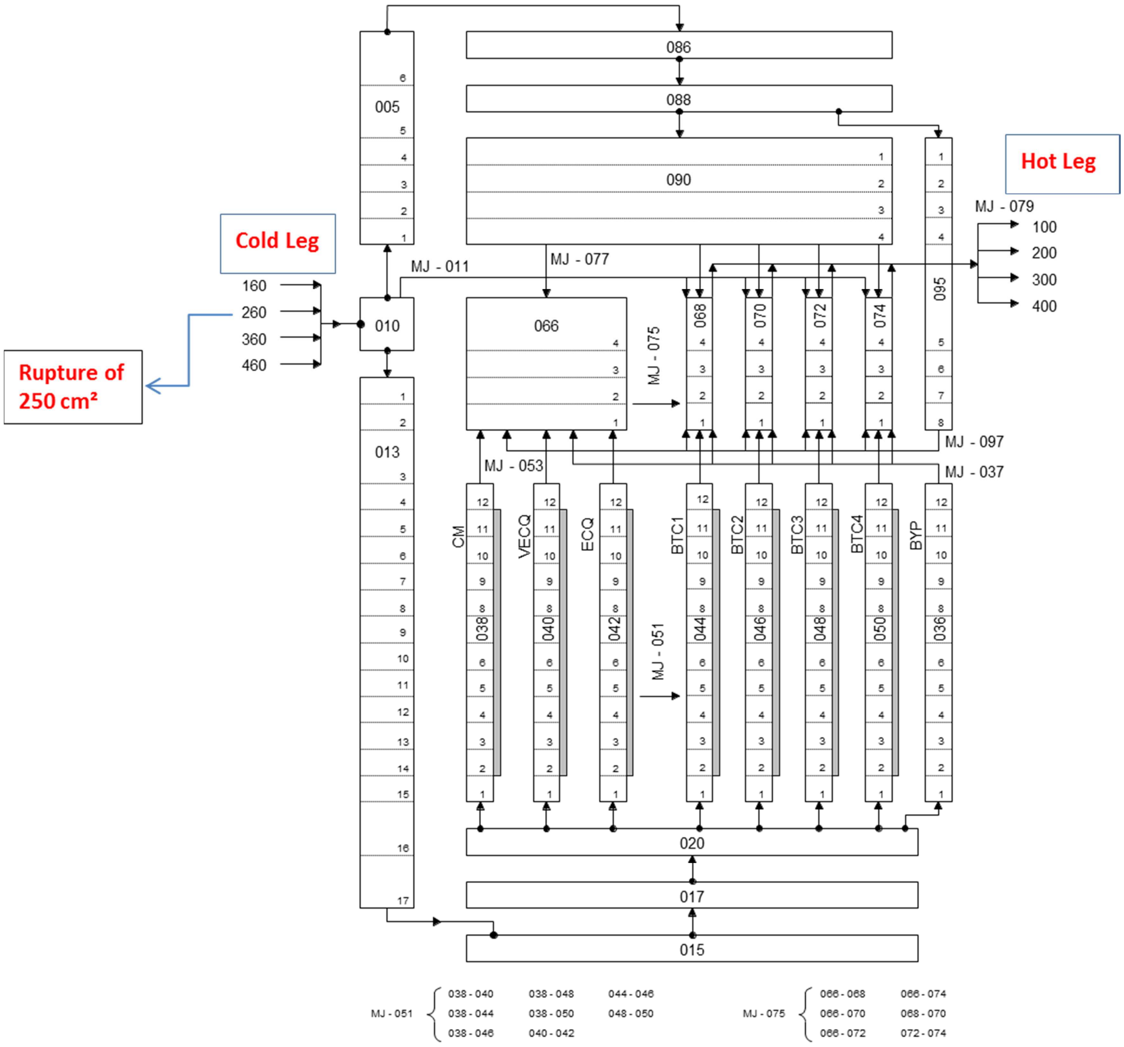

Figure 2: ANGRA 2 nuclear reactor core nodalization to RELAP5 code.

The RELAP5 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 2. 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 3 and 4 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 2: Flow regime number (RELAP5)

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 3: 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 4: Wall convection heat transfer mode numbers

Mode Number	Heat transfer phenomena	Correlations References
0	Noncondensable-steam- water	[5, 6, 7, 8, 9, 10].
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	[11].
4	Saturated nucleate boiling	Same as mode 3.
5	Subcooled transition boiling	[12].
6	Saturated transition boiling	Same as mode 5.
7	Subcooled film boiling	[5, 6, 7, 8, 9, 10, 13, 14].
8	Saturated film boiling	Same as mode 7.
9	Supercritical two-phase or single-phase gas	Same as mode 0.
10	Film wise condensation	[15, 16, 17].
11	Condensation in steam	Same as mode 10.
3 & 4	Horizontal bundles with nucleated boiling	[5, 18, 19].

3. RESULTS

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).

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 250 cm^2. This is the size of the rupture considered in this case. 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.

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.

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.

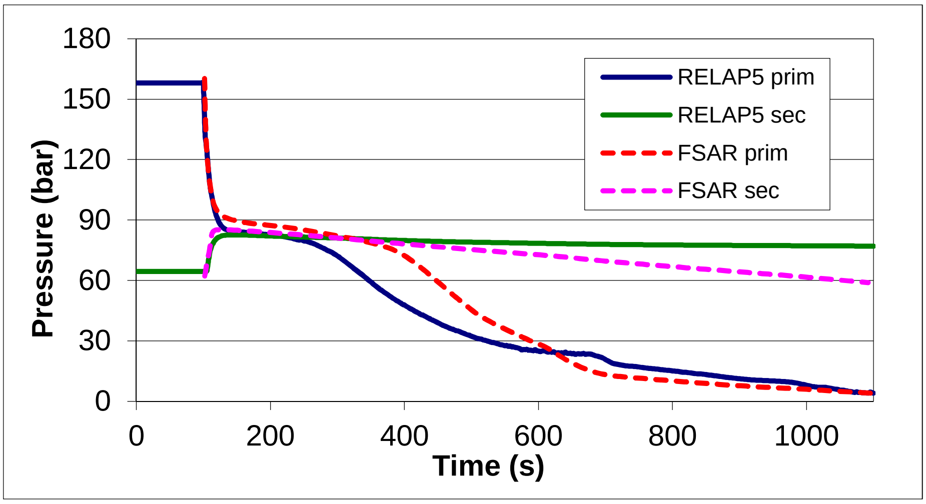

Figure 3: Pressure in the primary and secondary loops of ANGRA 2 (RELAP5 and FSAR-A2).

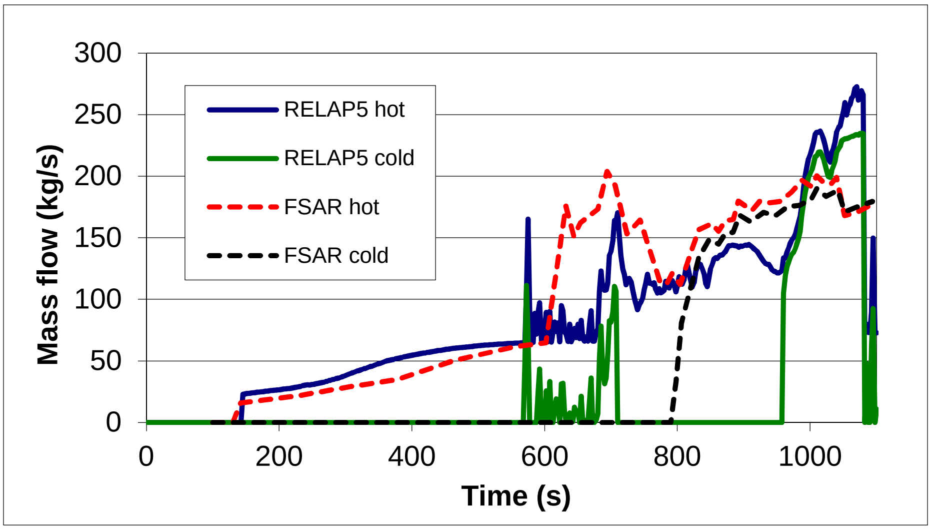

Figure 4: Mass flow in the lines of ECCS – Loops 10 and 20 (RELAP5 and FSAR-A2).

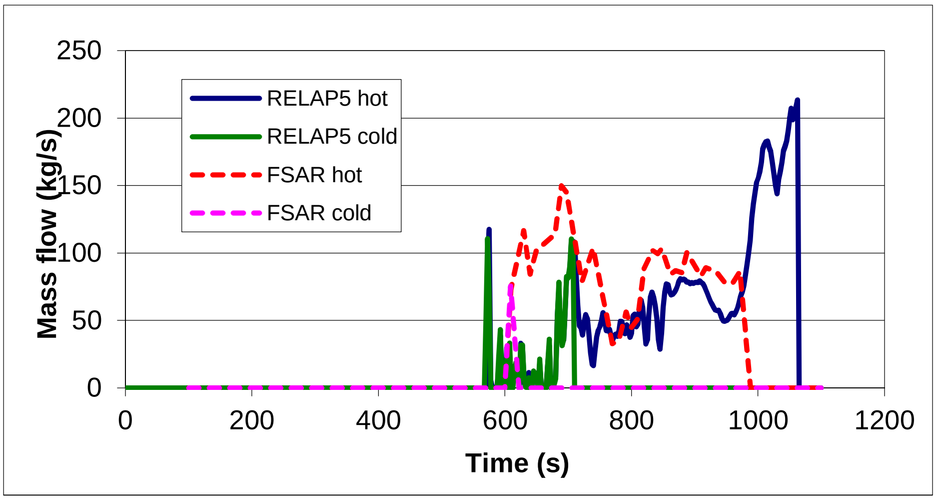

Figure 5: Mass flow in the lines of ECCS – Loops 30 and 40 (RELAP5 and FSAR-A2).

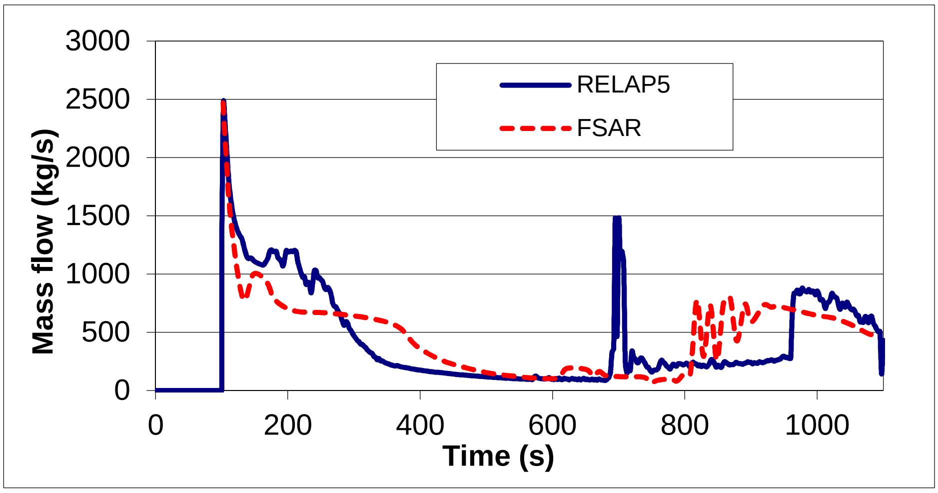

Figure 6: Mass flow in the break (RELAP5 and FSAR-A2).

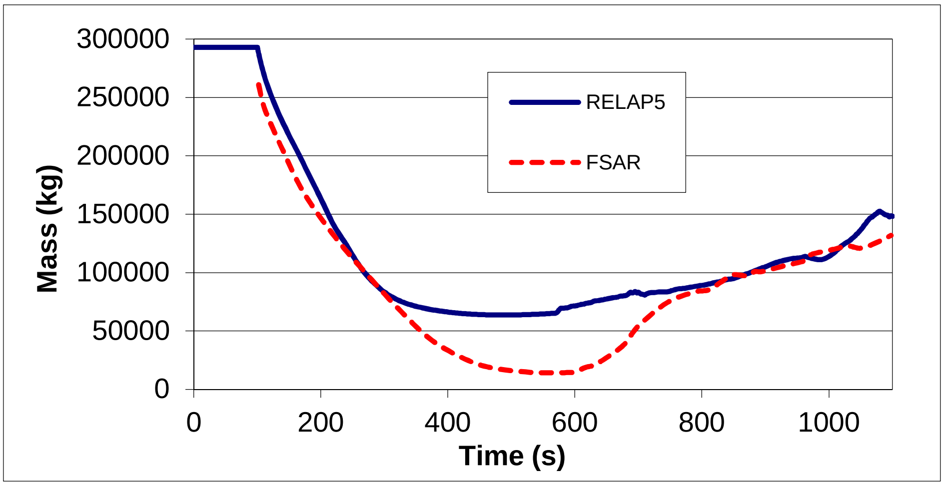

Figure 7: Primary coolant mass inventory (RELAP5 and FSAR-A2).

Figure 8 shows the void fraction in the rupture. Between 350 and 670 seconds of simulation, only vapor in the rupture was observed to RELAP5 simulation. Note that between 260 until 680 seconds the void fraction to RELAP5 simulation is higher than FSAR-A2 one, therefor the core void fraction and the temperatures are higher to RELAP5, too.

Figure 9 shows void fraction in the upper region of the hot channel of the core of ANGRA 2. Between 225 and 425 seconds, only vapor as observed to RELAP5 simulation. There aren´t FSAR-A2 core void fraction data.

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 2 and 4.

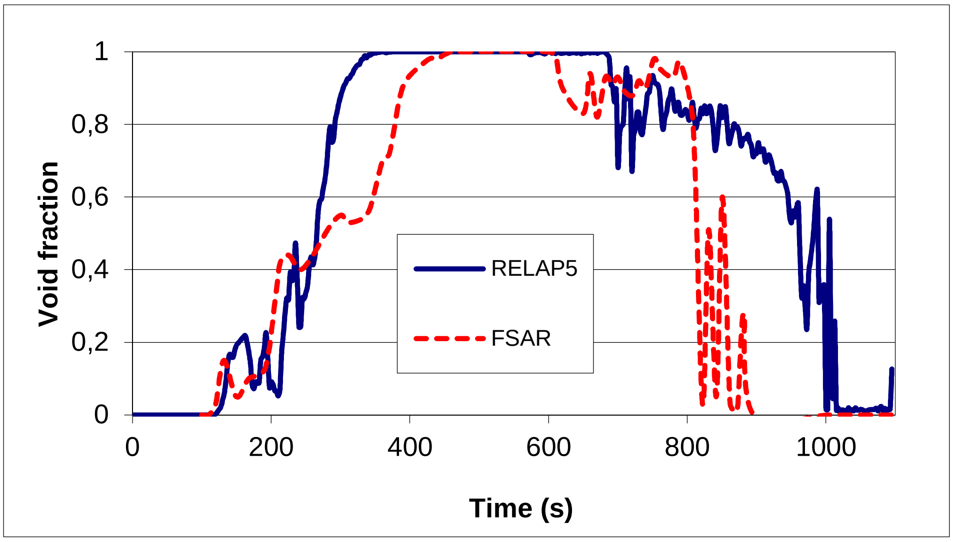

Figure 8: Void fraction in the break (RELAP5 and FSAR-A2).

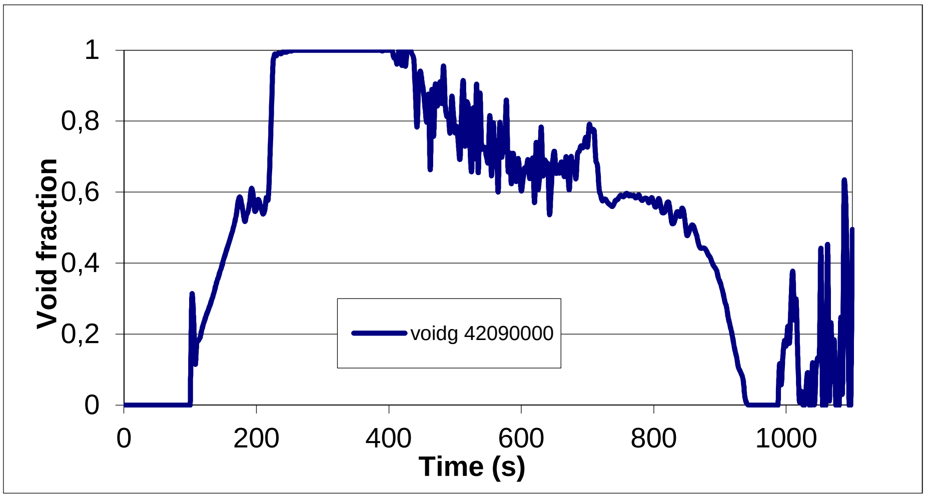

Figure 9: Void fraction to hot channel core of ANGRA 2 (RELAP5).

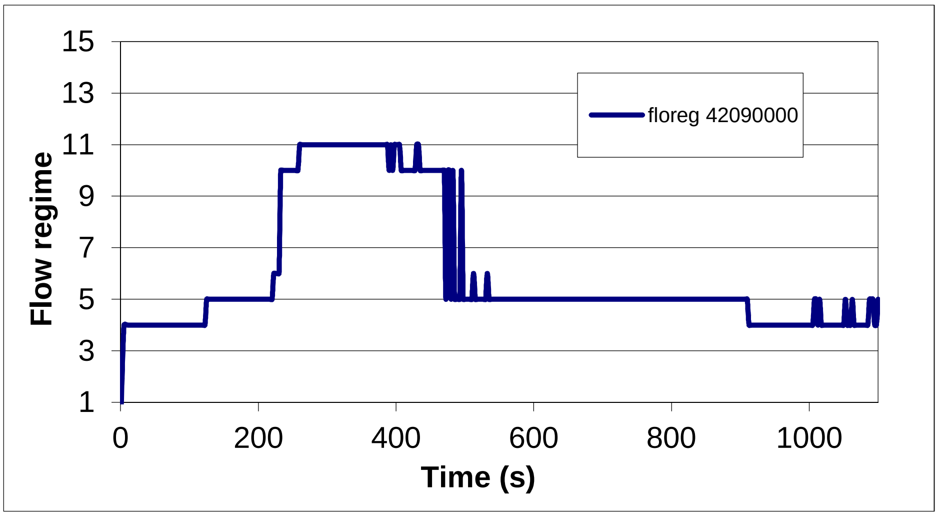

Figure 10: Flow regimes to hot channel core of ANGRA 2 (RELAP5).

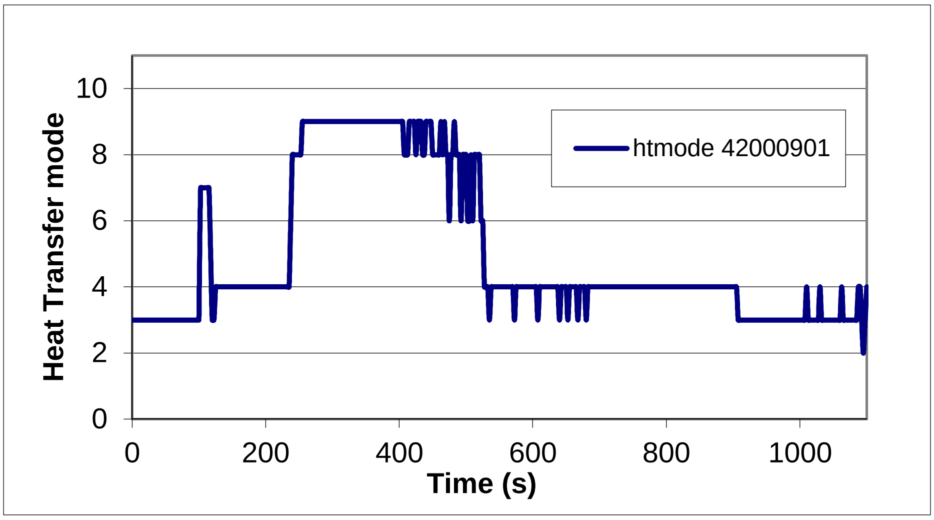

Figure 11: Heat transfer modes to hot rod cladding core of ANGRA 2 (RELAP5).

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 900 °C.

The ECCS system worked properly maintaining the integrity of the ANGRA 2 reactor core.

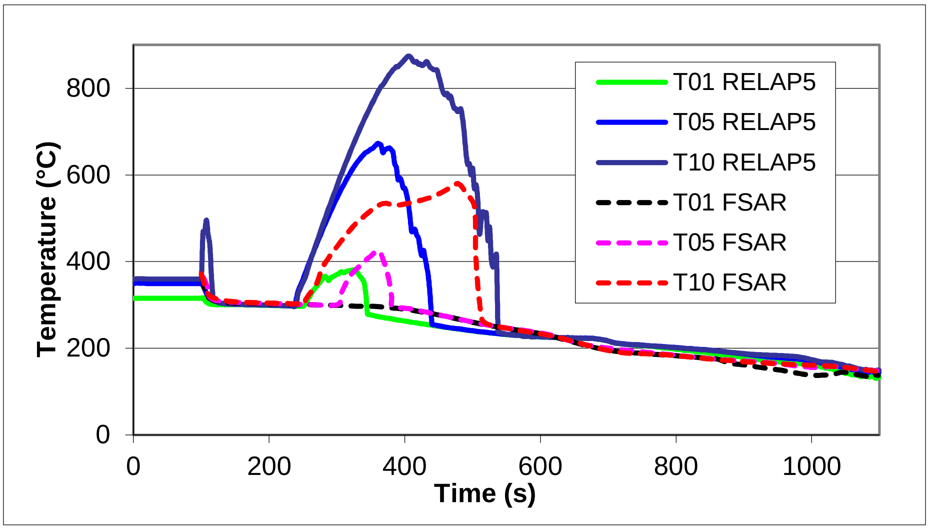

Figure 12: Hot rod cladding temperature of ANGRA 2 core (RELAP5 and FSAR-A2).

4. 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 250cm^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. 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*, Santos, SP, Brasil, 11-16 de agosto (2002).
5. ESDU Engineering Science Data Unit, 73031, *ESDU International Plc*, 27 Corshan Street, London, N1 6UA, (1973).
6. Kays, W.M. “Numerical Solution for Laminar Flow Heat Transfer in Circular Tubes”, *Transactions, American Society of Mechanical Engineers*, *77*, pp. 1265-1274, (1955).
7. 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).
8. Shah, M.M. *Heat Transfer and Fluid Flow Data Books,* Genium Publishing, Section 507.6 page 7, (1992).
9. 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).
10. McAdams, W.H. *Heat Transmission*, 3rd edition, New York: McGraw-Hill, (1954).
11. Chen, J.C. “A Correlation for Boiling Heat Transfer to Saturated Fluids in Convective Flow”, *Process Design and Development, 5,* pp. 322-327, (1966).
12. Chen, J.C.; Sundaram, R.K. and Ozkaynak, F.T. *A Phenomenological Correlation for Post-CHF Heat Transfer*, NUREG-0237, (1977).
13. Bromley, L.A. “Heat Transfer in Stable Film Boiling”, *Chemical Engineering Progress, 46*, pp. 221-227, (1950).
14. 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).
15. Nusselt, W. “Die Oberflachenkondensation des Wasserdampfes”, *Ver. deutsch. Ing., 60,* (1916).
16. 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).
17. 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).
18. Forster, H.K. and Zuber, N. “Dynamics of Vapor Bubbles and Boiling Heat Transfer”, *AIChE Journal, 1,* No. 4, pp. 531-535, (1955).
19. 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).