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

RELAP5 CODE SIMULATION OF THE SMALL BREAK LOSS OF COOLANT ACCIDENT OF 80 cm² IN THE COLD LEG OF ANGRA2 PRIMARY LOOP

Eduardo M. Borges[1,2] Gaianê Sabundjian[1] Francisco A. Braz Filho[2] and Lamartine N. F. Guimarães[2]

[1] 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

[2] Instituto de Estudos Avançados - IEAv
Trevo Cel. Av. José Alberto Albano do Amarante 01
12228-001 São José dos Campos, SP
fbraz@ieav.cta.br
guimarae@ieav.cta.br

ABSTRACT

The aim of this paper was to simulate and evaluate the basic design accident of 80 cm² small break loss of coolant accident (SBLOCA) in the cold leg of the primary loop of the Angra2 nuclear power plant. In this simulation, it was verified that the actuation logics of the Angra2 Reactor Protection System (RPS) and the Emergency Core Cooling System (ECCS) used in this simulation worked correctly, maintaining core integrity with acceptable temperatures throughout the event. The results obtained were satisfactory when compared with those presented by the Angra2 Final Safety Analysis Report (FSAR/A2).

1. INTRODUCTION

The Loss-of-Coolant Accident (LOCA) is considered a design basic accident in Nuclear Power Plants (NPPs) and has been studied since the first nuclear accidents. After the accident at the Three-Mile Island (TMI), more attention to the safety analysis was concentrated on the Small Break LOCA (SBLOCA) as a likely accident deserving of detailed analyses [1].

This accident is simulated by RELAP5/MOD3.2.gama [2] code, and it consists of the partial break of the cold leg of the Angra2 nuclear power plant. The rupture is the 80 cm^2 in the cold leg of primary loop how described in detail in the Chapter 15 of the Final Safety Analysis Report of Angra2 (FSAR-A2) [3].

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 [2].

To mitigate the consequences of a postulated LOCA, the Emergency Core Cooling System (ECCS) is used. In Angra2 nuclear reactor, the ECCS water is injected into the cold and hot legs in the primary system. The SBLOCA is characterized by slow blow down in the primary loop, allowing the actuation of the ECCS when the water is introduced in the circuit.

Results presented in this paper showed the correct actuation of the ECCS guaranteeing the integrity of the Angra2 reactor core.

2. INITIAL AND BOUNDARY CONDITIONS

The adopted initial and boundary conditions used in this simulation are in agreement with the Final Safety Analysis Report of Angra2 (FSAR/A2) [3]. These conditions are general for all LOCA cases of FSAR/A2. The actuation set points of ECCS are given as input data for RELAP5 code [4]. Some of the main boundary conditions adopted in this simulation are:

- reactor operating at 100% power to simulate a LOCA. Table 1 shows initial conditions (in FSAR/A2 all LOCA analyzes were performed at 106% power, conservative condition);
- the reactor core at start condition cycle-to-cycle balance (i.e. 6–day full power operation and 0.2 MWd/kg burn);
- top rod axial profile of power (more power at the top);
- decay heat according to Table ANS79-1, with a multiplicative factor of 1.08;
- shutdown signal (scram) of the reactor: conservatively considered the second shutdown signal, disregarding the 1st sign;
- reactivity scram (for reactor shutdown): equivalent reactivity of all the control rods except for the most reactive rod; additionally, a delay of 0.2s is considered for initial control rods fall and 2s for complete fall;
- assumed the condition of Emergency Power Mode (EPM), i.e. availability of external power, occurring at the same moment of the reactor shutdown and turbine insulation;
- failure and repair criteria account for the diesel generator circuits 30 and 40, which causes the unavailability of high pressure injection pumps and residual heat removal ECCS connected to these circuits;
- shutdown (coastdown) of the cooling pumps and reactor due to the loss of external power;
- considered the secondary cooling at a rate of -100 K / h, when the primary pressure (PRCs) <13.2 MPa and the containment pcont> 0.103 MPa;
- The performance criteria of ECCS: 2 of 3 signals; pcont> 0.103 MPa; PRCS <11.0 MPa; pressurizer level (LPZR) <2.28 m;
- criteria for the High Pressure Injection Pump (HPIP): criteria reached, + 32s delay due to MPE, +5 s delay for pump start in;
- criteria for the Residual Heat Removal Pump (RHRP): PRCS <1.0 MPa, + 37s delay due to MPE + 5s delay to the start of the pump;
- criteria for providing auxiliary feedwater: steam generator level (LGV) <5m.

Table 1 shows initial conditions, to this simulation how in FSAR/A2 to all LOCA analyzes were performed at 106% power, this is more conservative condition.

Table 1.: Angra2 Steady State Conditions

PARAMETERS	UNIT	[RFAS/A2]	RELAP5	ERROR (%)	
				USED	MAX OK
Core Thermal Power	MW	3765	3768,4	0,09	2,0
Vessel Pressure Loss	bar	2,93	2,815	-3,92	10
Core Pressure Loss	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 Rate	kg/s	17672,0	17671,00	-0,01	2,0
Core Bypass Flow Rate	kg/s	846,00	845,69	-0,04	10,0
Hot Leg Bypass Coolant Flow Rate	kg/s	188,00	188,21	0,11	10,0
Head Vessel Coolant Flow Rate	kg/s	94,00	93,98	-0,02	10,0
Steam Generator					
Exit Steam Pressure	bar	64,5	64,50	0,0	0,1
Primary Pressure Loss	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 Flow Rate	kg/s	513,9	512,34	-0,30	2,0
Recirculation Flow Rate	kg/s	1541,7	1541,3	-0,03	10,0
Liquid Level	m	12,2	12,34	0,14 m	0,1 m
Thermal Power	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 Loop					
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
Loop Coolant Flow Rate	kg/s	4700,0	4699,70	-0,01	2,0
Total Loop Pressure Loss	bar	6,5	6,37	-2,00	10,0

3. THE NODALIZATION OF ANGRA2 USING THE RELAP5 CODE

Angra2 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 Angra2 nuclear power plant [5].

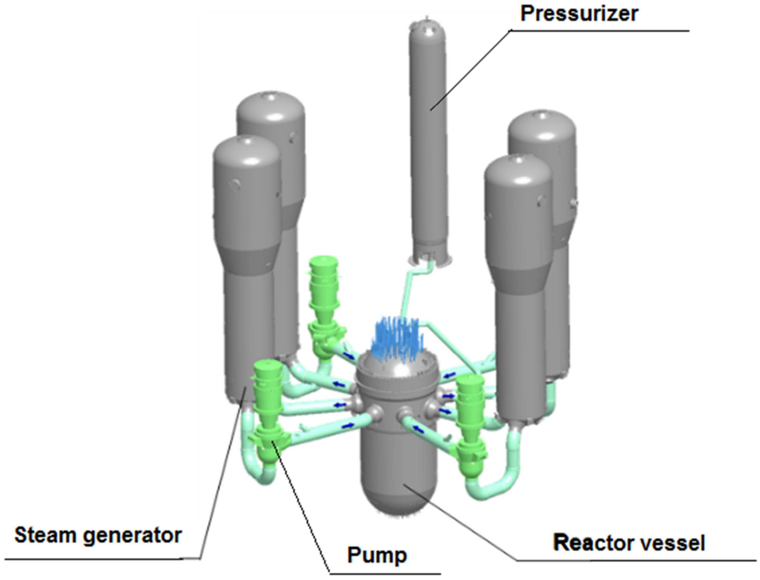

Figure 1: Arrangement of the Angra2 nuclear power plant components.

For each postulated LOCA, the ECCS performance is different. The Chapter 15 of the Final Safety Analysis Report of Angra2 (FSAR-A2) reports the ECCS actuation [3] 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 2.

Table 2.: 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 Angra2 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) [6, 7].

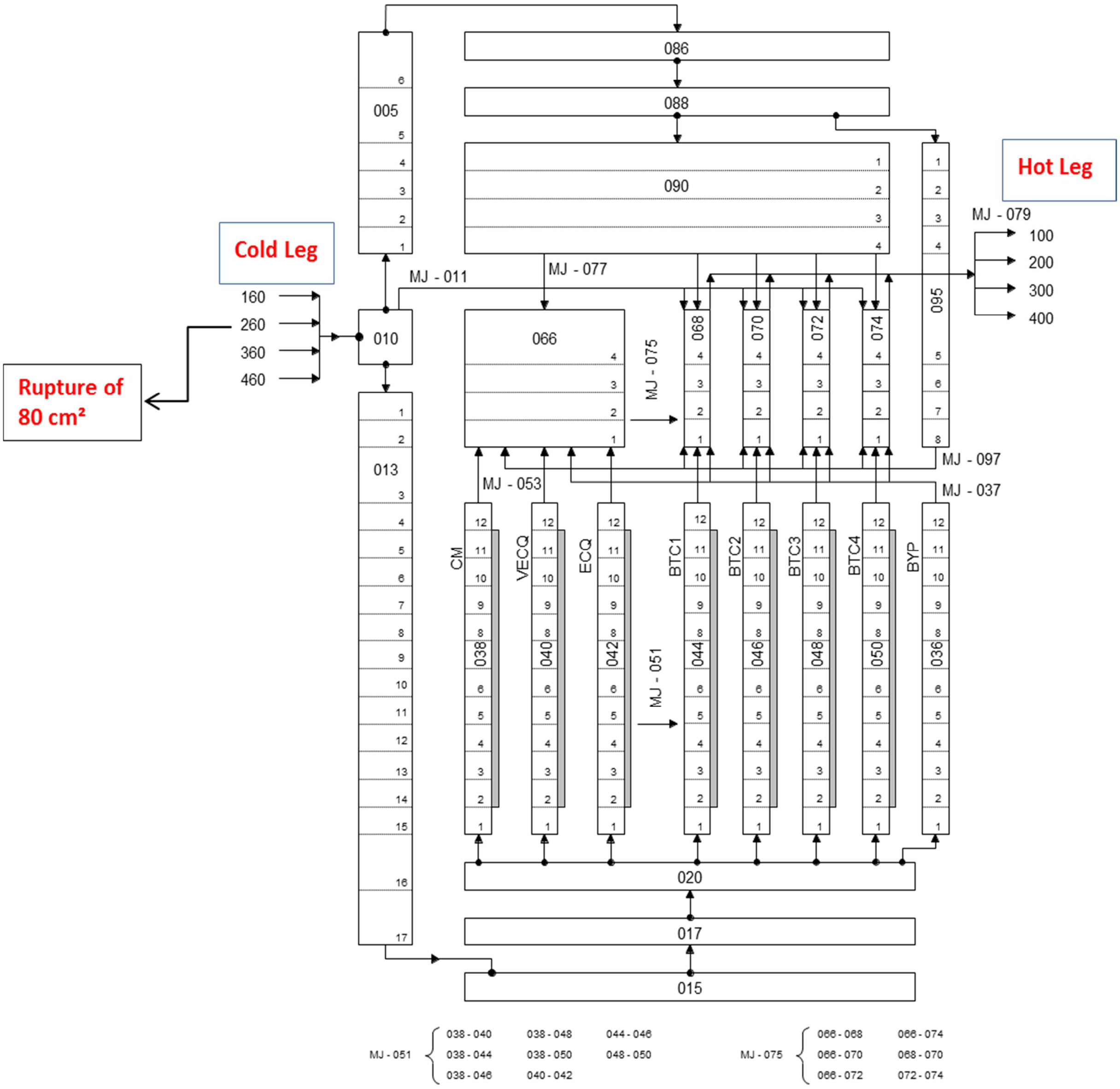

Figure 2: Angra2 nuclear reactor core nodalization to RELAP5 code.

4. RESULTS

The accident started after 100 seconds of the steady state simulation time, when the valve 951 is 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 80 cm^2. This is the size of the rupture considered in this case.

Figure 3 represents a single break (partial). The component 255 (PIPE) that is a piece of piping the cold leg of the primary circuit 20 Angra2 reactor, which will break; Component 960 (SINGVOL) represents containment; and component 951 (VALVE) the valve. This valve with passage area defined by the breaking size to be simulated, connecting the pipe broken to containment and is opened at the desired moment.

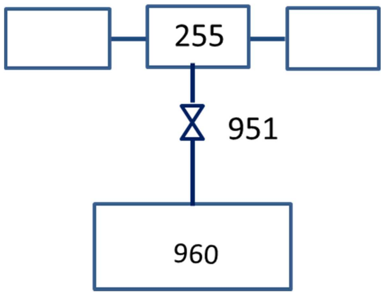

Figure 3: Single break simulation.

Table 3 provides a summary of the analyzed accident, the temporal sequence of operation and evaluation of Angra2 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.

Table 3.: SBLOCA 80 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.	115.1	216.5
100 K/h secondary-side cooldown (p_{RCS} < 132 bar and p_{cont} > 1,03 bar)	145.1	260.7
ECCS criteria met (p_{RCS} < 110 bar and p_{cont} > 1,03 bar)	125.4	223.4
Safety injection pumps start (High pressure pump start)	155.1	253.5
Accumulator injection starts	3232	3405
Hot channel recovered	2800	2900
Cold-leg accumulators isolated (500 s after ECCS criteria signal)	625.4	723.4
The end of Simulation	5000	5000

Figures 4 to 10 show the results obtained from SBLOCA of Angra2 analysis using RELAP5 code. These data 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 4 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. Can to be notes these data is very similar.

The ECCS system operates in function of the primary pressure, therefor the ECCS to RELAP5 code simulation is fast than FSAR-A2 one. Figures 5 and 6 show the mass flow of ECCS lines to RELAP5 and FSAR-A2. Notes that mass flow rate of ECCS cold line to loop 30 and 40 is zero all time.

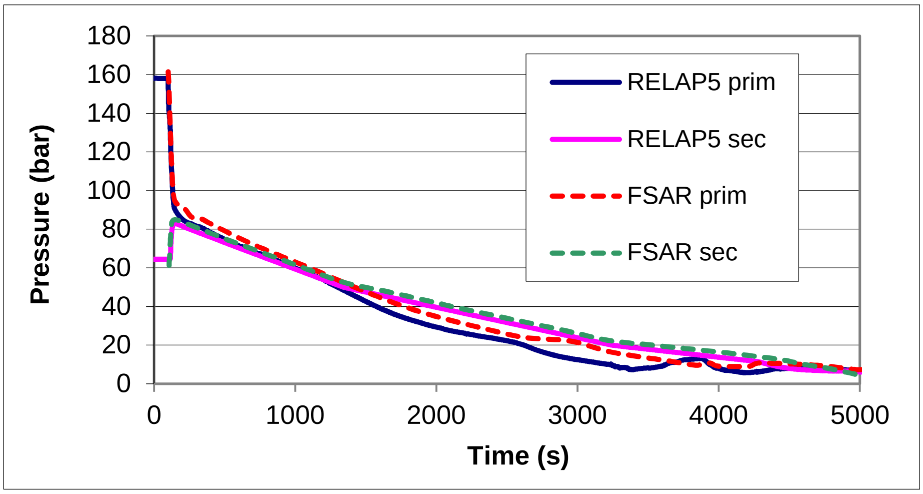

Figure 4: Pressure in the primary and secondary loops of Angra2.

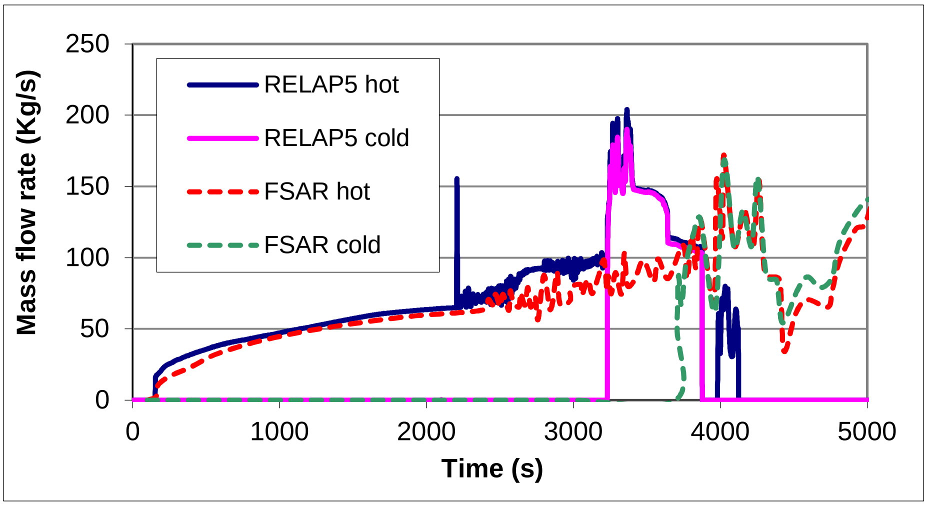

Figure 5: Mass flow in the lines of ECCS – Loops 10 and 20.

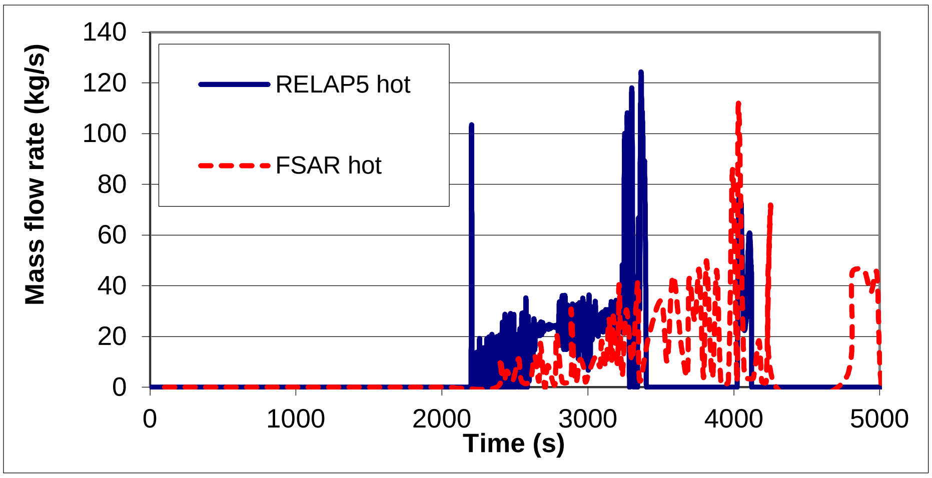

Figure 6: Mass flow in the lines of ECCS – Loops 30 and 40.

Figure 7 shows the mass flow rate in the rupture to RELAP5 and FSAR-A2. Can to be notes these data are very similar until 2200 seconds. After this time, the RELAP5 data are higher than FSAR-A2 one until 4000 seconds. Then FSAR-A2 data are higher RELAP5 simulation. To RELAP5 simulation the mass flow rate in the break data, after 4430 seconds are zero.

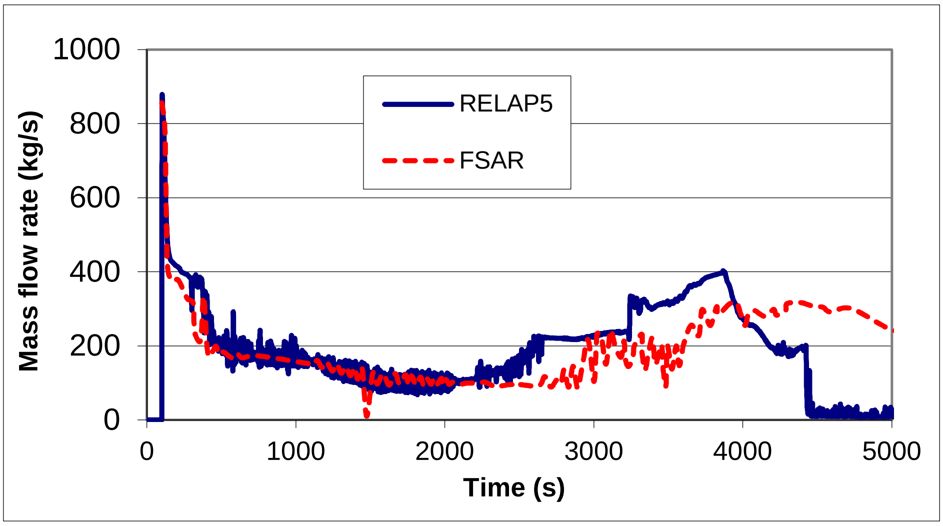

Figure 7: Mass flow in the break.

Figure 8 shows the primary loop coolant mass inventory. 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.

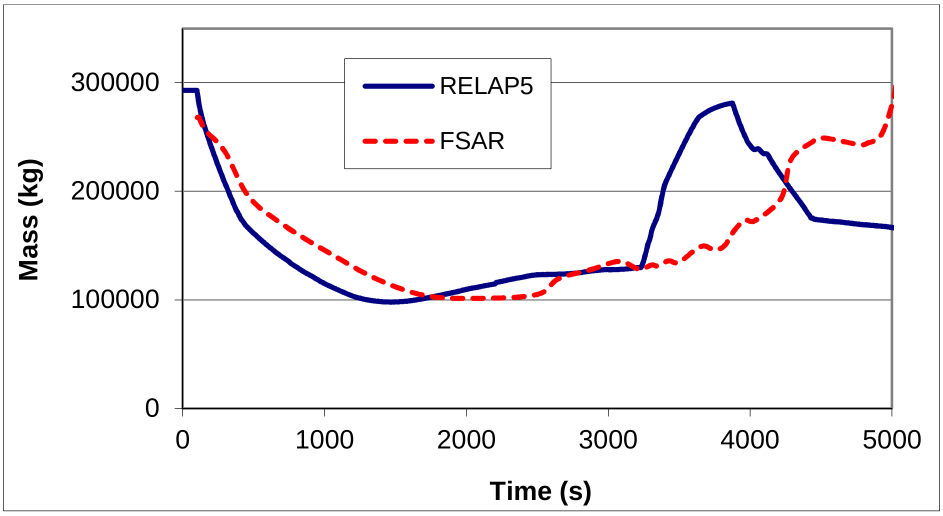

Figure 8: Primary coolant mass inventory.

Figure 9 shows the void fraction in the rupture to RELAP5 and FSAR-A2. Can to be notes these data are very similar until 2600 seconds, and the RELAP5 data are higher than FSAR-

A2 one. Then to FSAR-A2 there are some oscillations, and after 3600 seconds the void fraction in the rupture is zero, therefor there are only liquid water in the break to FSAR-A2. To RELAP5 simulation since 3100 seconds until 4430 seconds there are only liquid water in the break, but after 4430 seconds the void fraction in the rupture is one, therefor there are only steam water in the break to RELAP5 simulation.

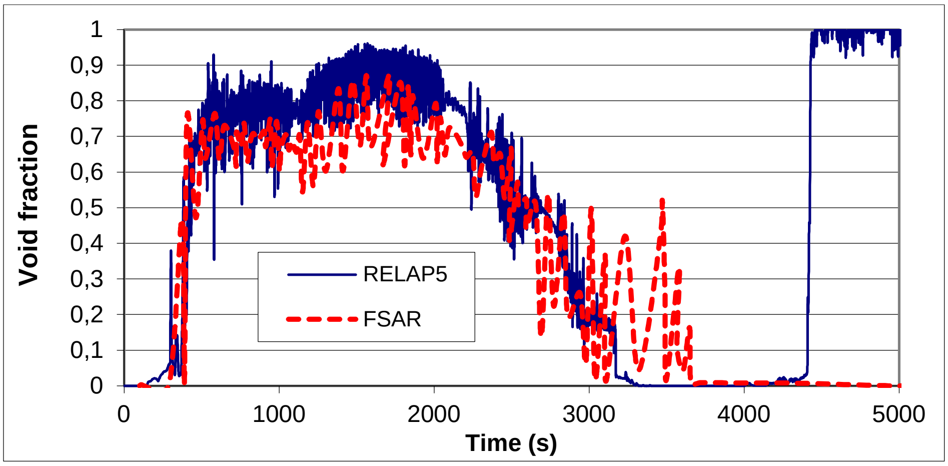

Figure 9: Void fraction in the break (RELAP5 and FSAR-A2).

Figure 10 shows cladding temperature of the top of hot rod of the core of Angra2 nuclear plant to RELAP5 simulation and FSAR-A2. Some these FSAR-A2 data are higher than RELAP5 one. But between 540 and 620 seconds there are oscillations in the hot rod core cladding temperature to RELAP5 simulation, and this temperatures are higher than FSAR-A2 one, but these data are lower than 370 °C, that is the initial cladding temperature of the top of hot rod of the core to FSAR-A2. And after 3830 seconds cladding temperatures to FSAR-A2 data are higher than RELAP5 one.

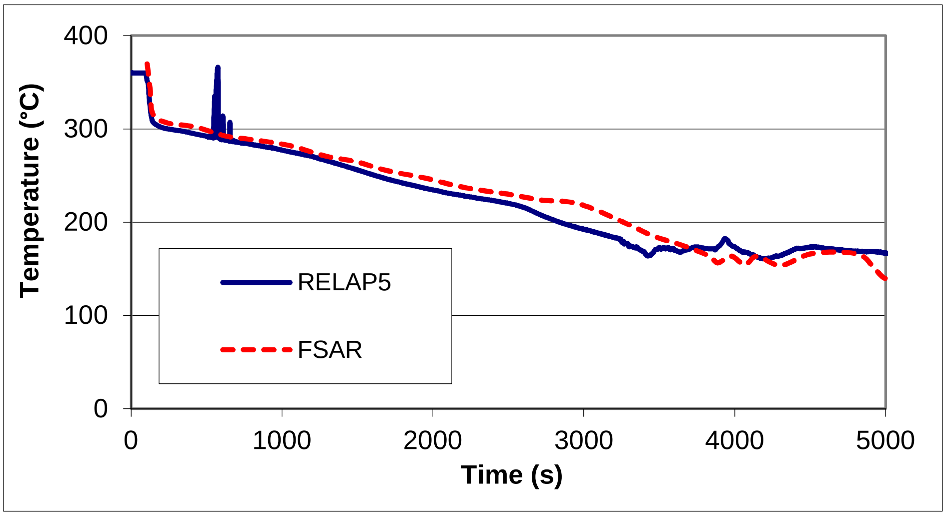

Figure 10: Hot rod cladding temperature of ANGRA2 core.

5. CONCLUSIONS

Can be notes that in RELAP5 code simulation, the primary pressure decreases faster than FSAR-A2 one. How the ECCS system operates in function of the primary pressure, therefor the ECCS to RELAP5 code simulation is fast than FSAR-A2 one. And the evaluation of the most important variables in this accident with FSAR-A2, when compared to their RELAP5 code simulation data one, showed that in this accident analysis FSAR-A2 was more conservative than the RELAP5 code.

Results presented in this paper showed that the actuation logics of the Angra2 Reactor Protection System (RPS) and the Emergency Core Cooling System (ECCS) used in this simulation worked correctly, maintaining integrity of Angra2 reactor core, with acceptable temperatures throughout the event.

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