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SIMULATION OF CHANNEL BLOCKAGE FOR THE IEA-R1 RESEARCH REACTOR USING RELAP5 / MOD 3

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ABSTRACT

Research reactors have great importance in the area of nuclear technology, such as radioisotope production, research in nuclear physics, development of new technologies and staff training for reactor operation. The IEA-R1 is a Brazilian research reactor type pool, located at the IPEN (Instituto de Pesquisas Energéticas e Nucleares). In this work is simulated with computer code RELAP5 / MOD 3.3.2 gamma, the effect caused by partial and complete blockage of a channel in MTR fuel element of the IEA-R1 core, in order to analyzed the thermal hydraulic parameters on adjacent channels.

1. INTRODUCTION

The RELAP5 is a highly known code, and calculates the cooling system behavior of a reactor, it can be used to simulate a wide variety of hydraulic systems and thermonuclear transients, involving steam mixtures, water, and uncondensable solutes. The MOD 3.4 version, which will be used in this work, has applications for research reactors [1]. The MOD 4.0 version is available for the computational development of the fuel behavior and specific cases of severe accidents.

The objective of this paper is to use the RELAP5 to analyze a possible severe accident scenario at the core of the Brazilian reactor research, the IEA-R1[2]. The phenomenology of the accident will be described for partial and total blockage at the inlet of the hottest fuel element of this reactor. The transient radial and axial temperature profile and power inside the hottest channel will be analyzed to predict the most important safety parameters in this. The initial knowledge of this distribution allows to monitor and prevent the instant when the channel starts undesired boiling crisis and, therefore, the occurrence of a severe accident. On the other hand, the initial nodalization can be enhanced and / or modified, providing the basis for modeling more complex and severe accidents.

2. RESEARCH REACTORS

Most of research reactors have a low potential for accidents when compared to power reactors. However, core configuration of this type of reactor are often changed, so some operational aspects need particular attention, especially as occupational workers are exposed to a high risk of accident. These modifications involve manipulating of fuel assemblies, control rods and experimental devices, many of which have substantial reactivity value. Also, laboratory devices are modified to suit the experimental requirements, and in these reactors visits are performed sites are performed visits directed to invited researchers, trainees, students and others who may have access to controlled areas with the active participation. Hence, it is possible to have nuclear interference, which could affect the neutron and thermal characteristics of the core and therefore its safety.

The reactor core modeling a reactor using computer codes must evaluate all operating states of the reactor have the capability to cool the reactor and maintaining the fuel in a thermally safe condition. Thus, the safety margins will be reliable to prevent or minimize damages to the fuel.

The input requirements for the RELAP5 can be divided into four parts:

• Hydrodynamic;
• Heat structures;
• Control systems;
• Nuclear.

The mathematical model of a hydrodynamic component is based on the mass conservation equations, energy and momentum to a fluid one / two-phase fluid contained in a tube. The RELAP5 offers the following hydrodynamic components: temporal volumes (TMDPVOL), single volumes (SNGLVOL), branches (BRANCH), tubes (PIPE), single junctions (SNGLJUN), temporal junctions (TMDPJUN) and valves (VLV). [1]

The input data required for modeling the hydrodynamic components are:

• All flow areas;
• All flow lengths;
• Vertical or horizontal guidelines;
• Geometric details to calculate hydraulic diameters;
• Roughness at the interface material wall-fluid;
• Information to calculate the flow variations (e.g., bend geometries, area expansion valve geometry);
• Initial conditions.

The adiabatic walls or a tube can be as a heat structures, therefore, the required input data of the structure are:

• Dimensions;
• Type of material;
• The thermal properties depending on temperature;
• The power distribution (if any);

• Surface roughness;
• The initial conditions of temperature;
• The cylindrical or cartesian coordinates;
• Geometry type (cylindrical, rectangular or spherical).

To run the RELAP5, it is necessary additional files to the code: the executable with ".exe" extension; tables of properties for light and heavy water, "tpfh2o", "tpfd2o" and DLL files in addition to the input data.

After collecting all the necessary information of the system being modeled, the next step is to isolate the main components and group them sequentially. And once defined the nodalization of the physical model, the next step is the division in discrete volumes, taking into account the Courant limit. The Courant limit is a numerical stability condition.

3. DESCRIPTION OF THE IEA-R1

The IEA-R1 reactor is a pool type research using fuel elements MTR. Each element has 18 fuel plates assembled in two side plates forming 17 independent closed channels. An overview of 5 MW reactor design parameters is presented in Umbehaum, 2000. [3]

The IEA-R1 reactor core has the shape of a parallelepiped being formed mainly by fuel elements, control elements, irradiation devices and reflectors. All elements and reflectors are vertically fitted into holes of a matrix plate, placed in a pool of demineralized water. The core cooling is made by downward forced circulation promoted by the main pump of the primary cooling circuit. The water, after passing through the core, is cooled in a heat exchanger and reinjected at the botton of the pool. The flow provided by the pump is divided among the various fuel elements and control, sample irradiators, secondary holes on the matrix plate, the channels between fuel elements and the channels between reflectors and radiators, branching out the collector region.

The MTR is the oldest fuel and many changes have been incorporated in its design to facilitate the research reactor operation and to meet the desired neutron flux requirement, the irradiation isotope production, etc. To ensure the stability and structural integrity, these fuels have a coating, generally consisting of aluminum. The MTR plate-type fuel can be divided into two regions: the fuel and the coating, a metallic sheath covering the fuel. The IEA-R1 standard fuel elements consist of 18 flat plates with an average thickness of 1.52 mm, arranged in a rectangular cross section of 76.1 mm x 79.76 mm.

The material of the standard fuel element analyzed is number 165, and as fuel U_3Si_2-Al element, the density 3.0g/cm³.

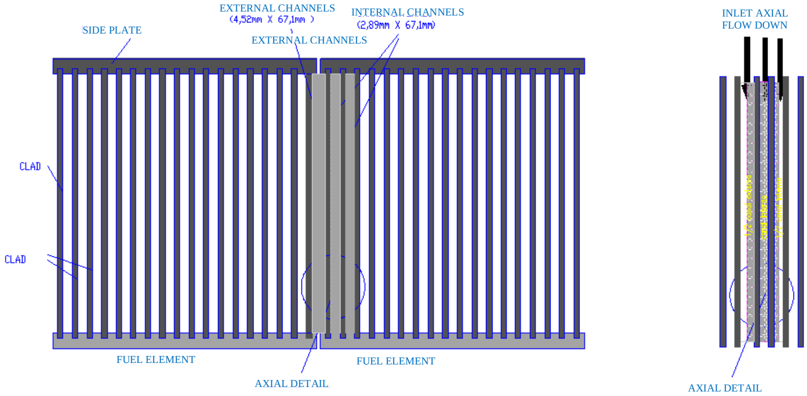


Figure 1: Typical section of two adjacent fuel elements IEA-R1. [2]

The cooling channel dimensions are 67.1 mm x 2.89 mm (thickness x width), with an overall height of 625 mm plate and the active height (heated area) of 600 mm. In the Fig. 2, a diagram of the plate type fuel element is shown.

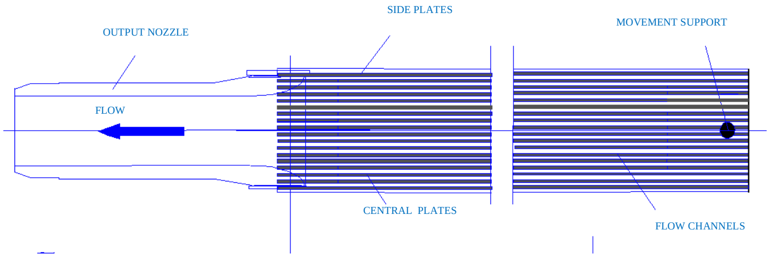
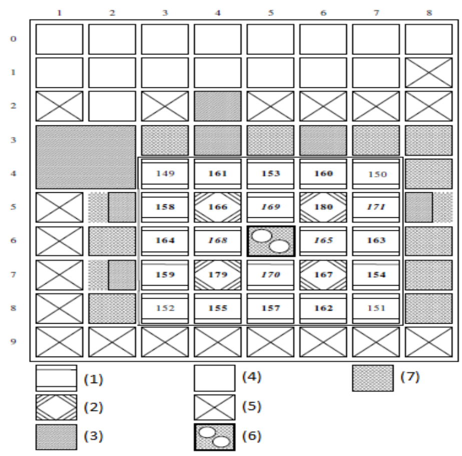


Figure 2: Element Standard IEA-R1 Reactor Fuel. [2]

4. BLOCK MODELING COOLING CHANNEL

This session is dedicated to the methodology proposed for the modeling of the blockage cooling channel in the nuclear reactor Brazilian, the IEA-R1. Figure 3 is the core configuration wherein patterned numbers are displayed to distinguish the elements.



0.647	0.845	0.919	0.939	0.743
0.886	0.980	1.401	1.140	1.286
1.032	1.397	0.000	1.482	1.120
0.922	1.084	1.391	1.039	0.959
0.663	0.821	0.878	0.836	0.675

Figure 3: Arrangement of the fuel elements and configuration of the IEA-R1 core where (1) standard fuel, (2) control elements, (3) irradiation device (4) plug (5) reflector graphite, (6) irradiation device and Beryllium (7) Beryllium reflector (left). Radial standard power distribution (right) [4].

Figure 3 shows the normalized radial power distribution, wherein the higher value of 1.482 corresponds to the standard fuel element 165.

The nodalization simulation proposed takes into account some considerations:

1. The IEA-R1 core consists of 20 standard fuel elements, as shown in Figure 3;
2. Each standard fuel element has 18 fuel plates;
3. Only 9 were simulated, 9 heat structures (HS), considering that the effect produced on the plate is symmetrical (Fig. 4);
4. The channel 165 is the hottest standard fuel element according to Figure 3;
5. Between the plates of the fuel element, there are cooling channels modeled as a pipeline (PIPE), and 8 having these structures.
6. To simulate a channel blockage or as any of it will be used 6 valves (VLV);
7. The pipes at the ends will be disregarded their blocking effects, due to the existing bypass;
8. The direction of the working fluid, the derivation of the components and the upper and the lower parts of the reactor pool.

Figure 4 shows the nodalization for a half fuel element of Figure 3, used to simulate the channel blockage.

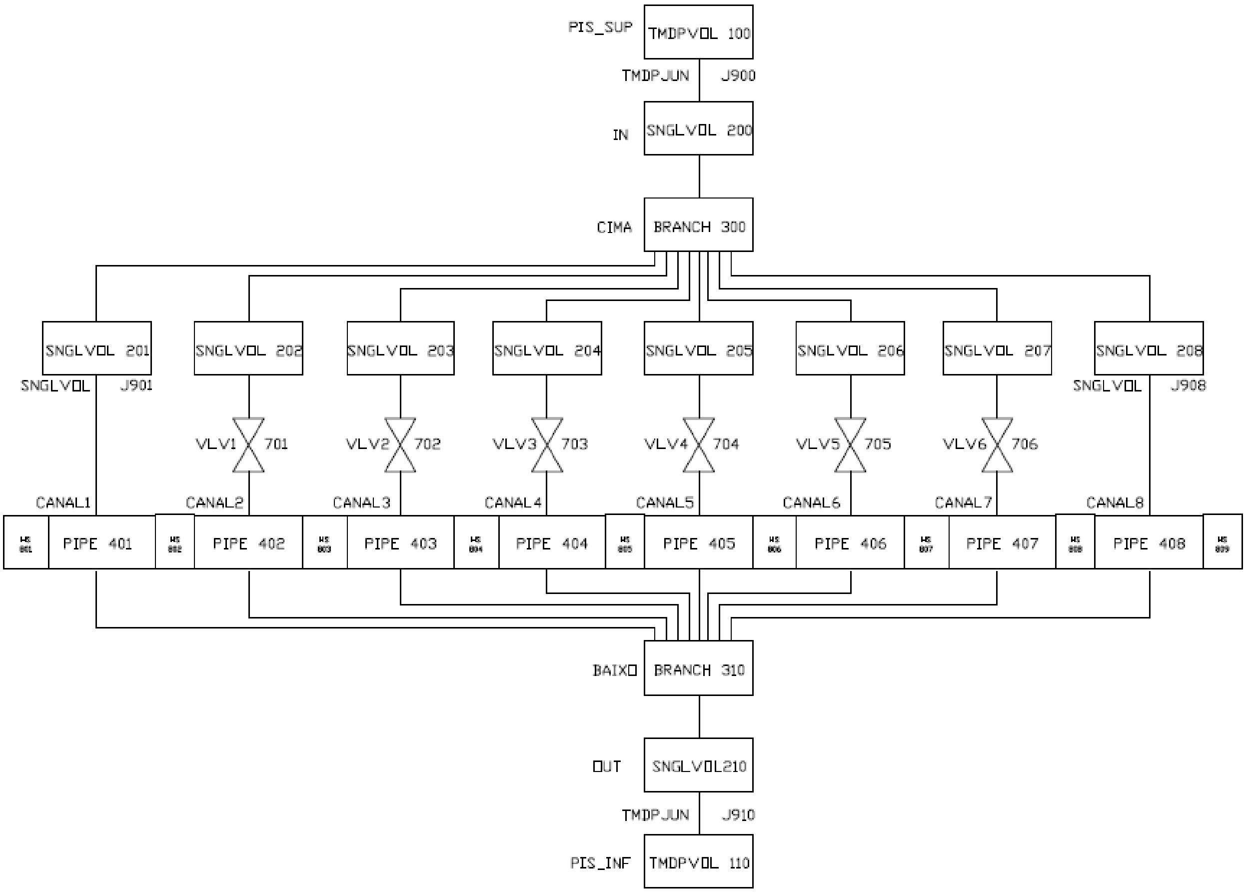


Figure 4: Nodalization proposed for the problem.

For the preparation of RELAP5 input data, it is necessary the reactor experimental data, reactor location and some calculations based on reactor data.

The control of the valves 701-706 is accomplished by trip command, that at times of 20, 40, 60, 80, 100 and 120 seconds respectively close valves 701, 702, 703, 704, 705 and 706 whereupon the reactor core partially blocking the channels between 20 and 100 seconds and 120 seconds in total blockade, as shown in Figure 6. After these steps the lock channels, thermal-hydraulics variables were analyzed.

The command and the opening and closing of the valve formats are described in Figure 5, which can be observed trip cards used in data entry simulation proposal and the expected result.

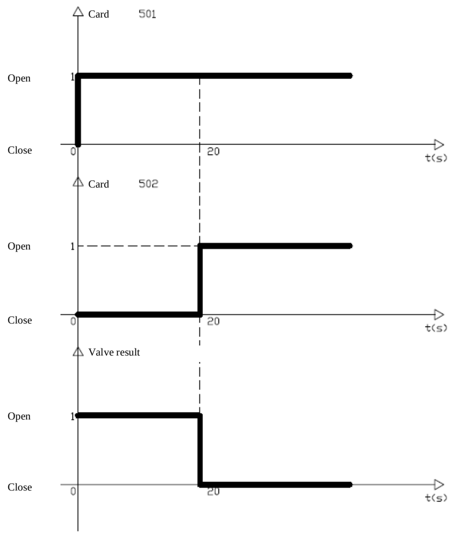


Figure 5: Logical states of the Boolean variables of the trip cards.

The expected result is that the shorter or equal to 20 seconds, the valve remains open, and after this time the valve to close. In RELAP5 at 60X card you can operate the parameters with the commands AND, OR and XOR. The truth table of each operation is given in the figure below, in which A and B are the input variables R and the result output.

Table 1: Table truth of logical variables AND, OR and XOR.

AND			OR			XOR		
A	B	R	A	B	R	A	B	R
0	0	0	0	0	0	0	0	0
0	1	0	0	1	1	0	1	1
1	0	0	1	0	1	1	0	1
1	1	1	1	1	1	1	1	0

In the case studied, it must use a logical variable whose inputs and outputs comply with the three criteria below:

1) A = 1, B = 0 and R = 1 (one closed and open valve tripping);
2) A = 0, B = 1 and R = 1 (one closed and open valve tripping);
3) A = 1, B = 1 and R = 0 (two trips open and valve closed).

Table 1 contains the truth tables of logical variables AND, OR and XOR and the command necessary for operation following the three criteria is the XOR command. For this reason, the card 601 is used this logic operation.

The trip 508 serves to signal 20 MPa pressure limit, the central channel, which contains the more critical temperature profile, and then actuate the scram. In this pressure limit, negative reactivity is inserted, can be seen in Figure 14.

Figure 6 shows the channel being blocked at the times previously reported. Note that when you close a whole channel, the mass flow is distributed in adjacent channels, until it reached 120 seconds, causing the water to flow only by the bypass, modeled on the pipes 401 and 408.

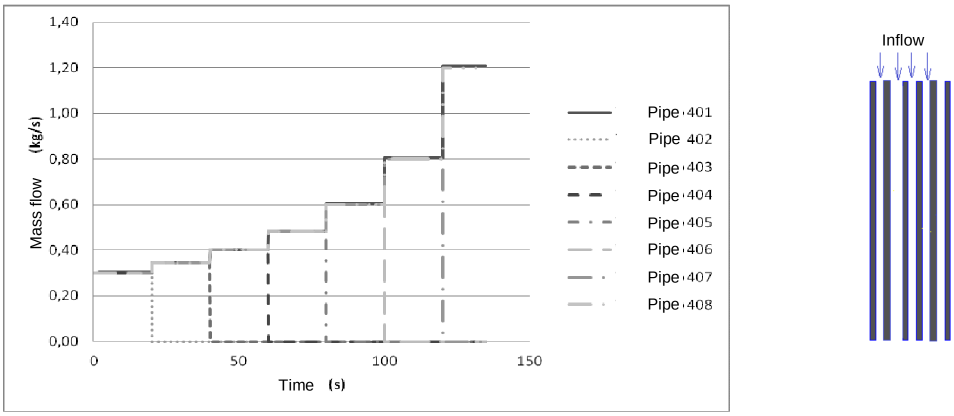


Figure 6: Mass flow (mflowj) the channels

For the various cards RELAP5, we should mention some important variables and how they were calculated, as well as the considerations.

5. RESULTS AND DISCUSSION

The channel blockage is a postulated accident in research reactors and can occurs when foreign materials are accidentally introduced into the channel and must have a low probability of occurrence since it may result in triggering event melt one of the fuel elements and the undesired release of radioactive materials to the cooling water. The frequency of occurrence of such an accident similar reactors in the IEA-R1 was the order of 10^{-2}/year for the Greek reactor and $1,3x10^{-5}$/year for the Australian reactor [5] When caught early release fission products, automatic shutdown of the reactor should be activated.

In the simulation with RELAP5 were modeled, because of the symmetry, the partial blockage scenarios six cooling channels, their respective fuels, adjacent plate type in hottest element and two bypasses (external channel). For this, used were artificially six trip type valves, hydrodynamic component RELAP5, which are sequentially closed for times of 20, 40, 60, 80, 100 and 120 seconds to block 100% of mass flow throughout the hottest fuel element. The logic operation of the valves was explained in the previous session.

Initially in each individual cooling channel flowing a volumetric flow rate of 19.5 m^3/h, which corresponds to a mass flow rate of approximately 0.3 kg/s. The first channel was 100%

blocked in the first 20 seconds. Figure 7 shows the transient that simulates closing the first valve (cooling channel block).

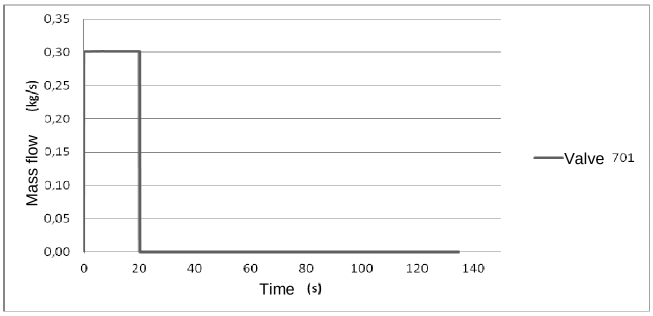


Figure 7: Transitional mass flow rate of the first cooling channel.

Subsequent blocks were performed at the times of 60, 80, 100 and 120 seconds, as shown in Figure 8. From the figure, it is observed that the refrigerant flow in the first channel is interrupted redistributed on subsequent channels and the bypass, which caused an increase in the mass flow 0.05 kg / s on each channel. Thus, after 20 seconds, the increased mass flow 0.35 kg / s on adjacent channels. While the former is kept locked, the second channel was 100% blocked for 40 seconds in this way the mass flow channels subsequent increased to 0.4 kg / s. A sequence of six closing valves continued until completely block the flow in the fuel element.

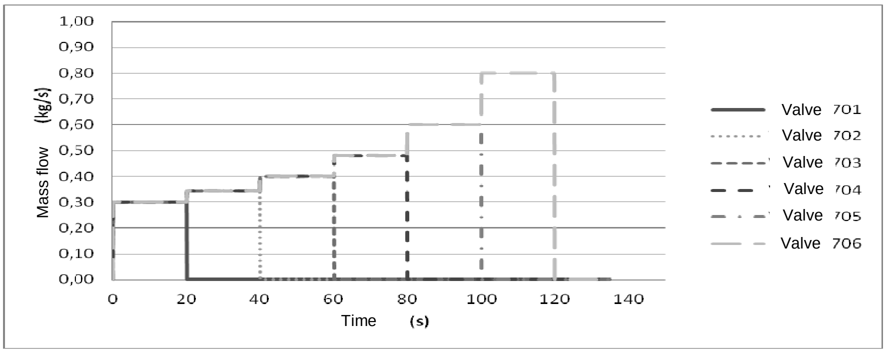


Figure 8: Transitional of mass flow in all blocked cooling channels.

Figure 9 shows the transient mass flow performed when the last valve block. In this case, all the cooling water bypasses the two flows of the fuel element. The two bypasses were modeled using the hydrodynamic component PIPE. The interactions between the individual channels causes the total mass flow flows through the bypass stepwise reached 1.3 kg/s.

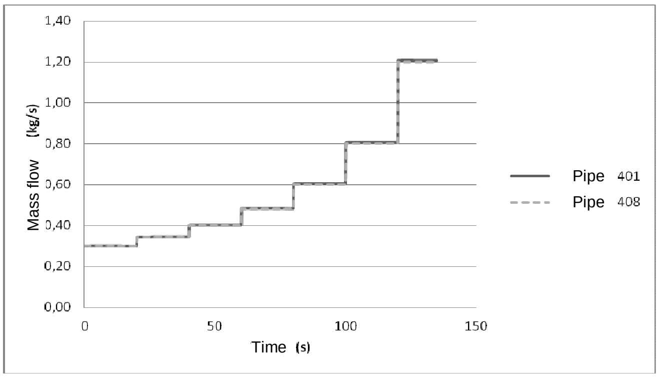


Figure 9: Transient mass flow of the bypasses

5.1. Results without scram

The scram is a procedure within the nuclear reactor used to force the reactor shutdown and/or decrease the neutron rate. In this initial simulation, the effect of scram was not included, for study the dynamic behavior of the fuel element without protection.

The 5 MW of total core power were divided among the 48 fuel elements control and 360 standard fuel elements. In figure 10 is shown the behavior of similar potency to each plate of the fuel element remains in the steady state at all locks simulated scenarios. The thermal power is due to the fission fuel and emission of radiation. The variable “rkfipow” is due to fission thermal power, 11.4 kW; “rkgapow” is the power due to the decay of the gamma emission, 0.86 kW, and finally “rktpow” represents the total amount of power for each plate contained in a fuel element MTR 12.26 kW. These results were modeled with the inclusion of the point kinetic card that RELAP available for version 3.4. For the modeling were assumed values of the neutron fraction of delayed βeff = 0.007693 and the generation time of the instantaneous neutron ʌ = 57.9 seconds.

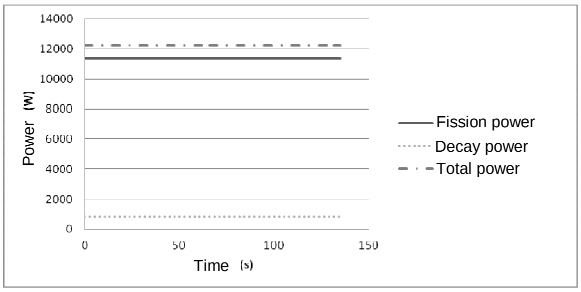


Figure 10: Power on each fuel element plate.

The pressures at the top and the reactor core base was simulated and maintained constant using the hydrodynamic component with TMDPVOL 100kPa values for TMDPVOL 10001000000 and 174 kPa in TMDPVOL 11001000000. Figure 11, left side, reproduces the transient these two points of the reactor pool. However, the transient pressure inside the cooling channels in all the hydrodynamic components in the lock channel scenario is shown in Figure 11, right. It is observed a profile with a growth rate variable for the pressure reaches values that exceed the critical pressure of water (Pcr = 22.04 MPa) to simulate proposal valves 6 and lower the critical water pressure when used only one valve closed, as commonly found in the literature. However, the content within the channels remains in the thermodynamic state of compressed fluid according to the numerical value of the static quality (vapor mass fraction) during the transition shown in Figure 12, left side.

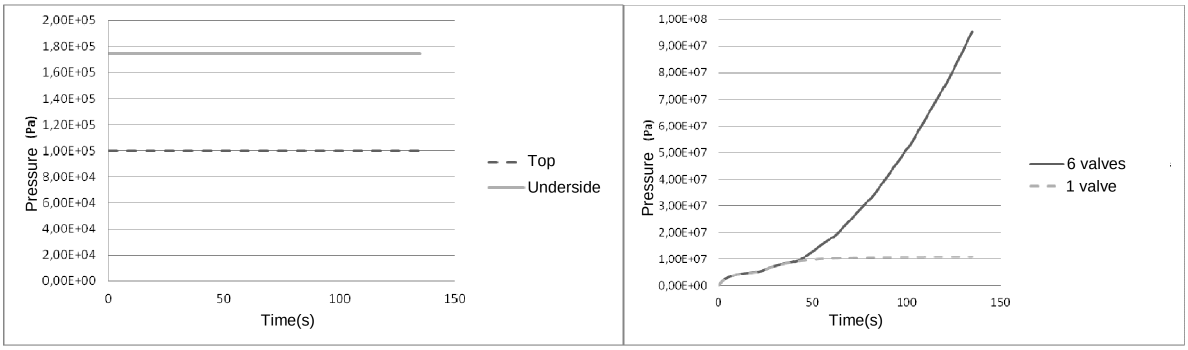


Figure 11: Transient pressure on the top and bottom of the fuel element (left) and pressure in the coolant channels using the valves 1 and 6 (right).

The quality of the static volume of coolant is crucial to check that there is no presence of boiling both within channels (Equals 401010000-408010000) and the system.

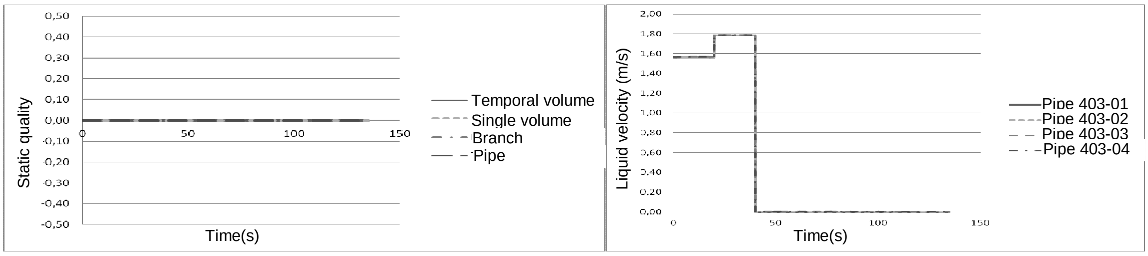


Figure 12: Static quality (left) and transitional speed of the liquid in the hot channel (right).

The values of the velocities for each phase of the channel 403, the hottest the element, and the third channel from the left to the right, confirms that subcooled liquid is not flowing which causes an increase in temperature, as is shown in figure 12, right. Before the third channel blocking in 40 seconds, the initial velocity in all axial we started in 1.5638m/s. By blocking the first and the second valve, the fluid velocity increased to approximately 1.789m/s.

Considering that each cooling channel is thermally connected by fuel adapters to power generation, the temperature in each cooling channel are affected. The RELAP provides the HEAT STRUCTURE component model that allowed the MTR fuel. Figure 13, left side, shows the transition temperatures of the fuel plates. The bypasses are always in touch with the refrigerant, so temperatures values do not change significantly until the total blocking. The maximum temperature difference was left in 16,29ºC bypass 801 (time 20 seconds), the maximum temperature difference was in the right 7,73ºC bypass 809 (time 100 seconds), see Figure 13, left side.

Figure 13, right side, shows the transient temperature of fuel at the internal plates at the nearest location and greater fuel. The temperatures values change significantly as the valves are closing sequentially until the complete block. The temperature difference reached 160,69ºC fuel in the fourth plate 804, followed by the fuel plate 803. Between these plates is the cooling channel 403, the hotter.

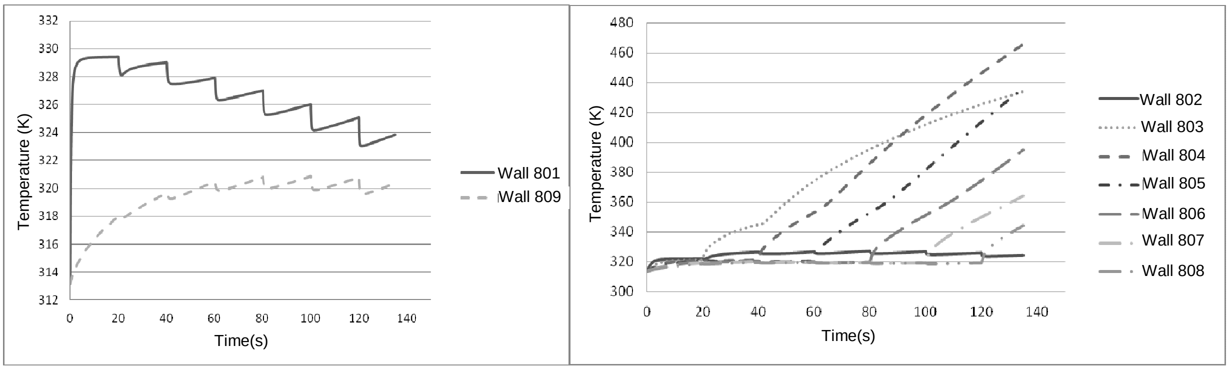


Figure 13: Temperature Transient fuel in bypass of the left plate 801 and right 809 (left side) and transient temperatures of the fuel element in all the plates (right side).

5.2. Results with scram

The introduction of a scram ensures reduction in reactor power in an emergency. This procedure is simulated with the external introduction of negative reactivity when the pressure of the hottest channel, 403 exceeded 20 MPa. The result of transient pressure and reactivity is shown in Figures 14 and 15, left side. The scram occurs at 63.8 s transient. Within the first 60 s. the reactivity of the system is 0 $, stable reactor. The reactivity was introduced as Table 2 or Figure 14.

Table 2: Reactivity inserted into the scram

Tempo(s)	Reatividade ($)
-1	0
0	0
10	-2
120	-4
140	-6
160	-8
180	-10
200	-12
220	-14
240	-16

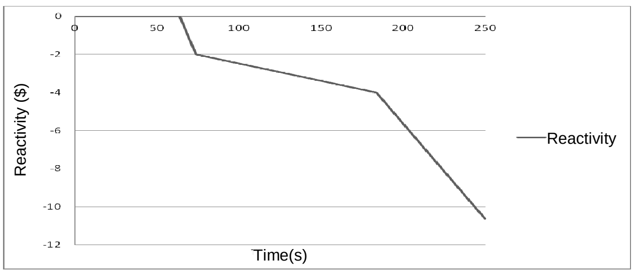


Figure 14: Transient inserted external reactivity in the IEA-R1.

Thus, the transient pressure to said scram forced to slow temporal decrease, see Figure 15, left side. The result of this little disruption affected the power due to the decay of the emission range (rkgapow) and significantly affected the power due to fission (rkfipow). This power of the sum generated by gamma emission with the fission, is the total power value (rktpow) and is subject to the forced shutdown of the nuclear reactor, see Figure 15, right side.

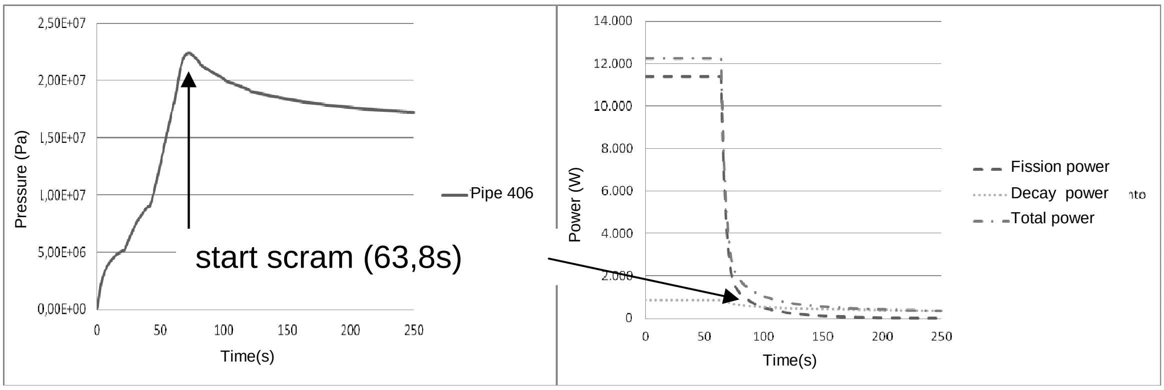


Figure 15: Pressure all hydrodynamic components, scram (left side) and transient power in each fuel element plate with scram (right side).

Finally, the axial temperature behavior is presented in the hottest fuel which reaches the maximum value of 365 K, below the temperature which affects the coating integrity, see figure 16.

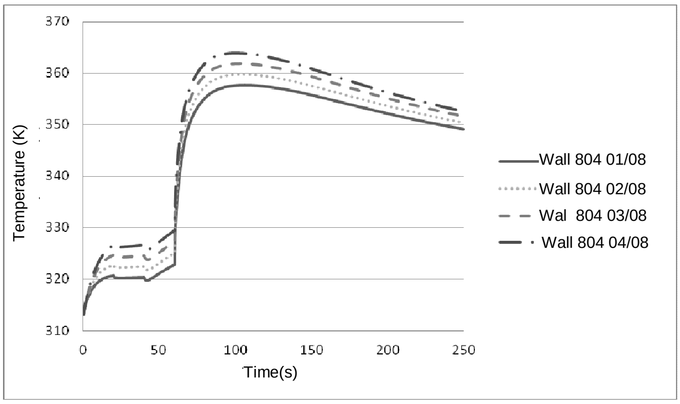


Figure 16: Axial Temperatures in the farthest part of the fuel, with scram.

6. CONCLUSIONS

The RELAP5 code, the most mature of thermal-hydraulic codes used in pressurized power reactors, was used to simulate a hypothetical accident project based on research nuclear reactor safety related. The phenomenology of partial block and the total warmer cooling channel, with and without scram, was analyzed for the Brazilian IEA-R1. The nodalização type employed for fuel MTR shown to be applicable to the situation lock patterned once the temperature has been shown to results qualitatively consistent with the analyzed data in the open literature and similar configuration to the reactors R1-IEA. Of course, no data experimental results can not validate the results quantitatively, for obvious safety reasons. However, the qualitative analysis shows that blocking the refrigerant to 6 channels in the hottest element does not affect the integrity of this type of reactor, no presence of boiling crisis; and that the extreme case, as analyzed, the response of the scram ensures safe shutdown of the reactor.

For the future, once reached the level of maturity as a user of this code is intended to simulate other accidents of this type of reactor, including nodalization of all core components.

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