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Research Article

Simulation of a TRIGA Reactor Core Blockage Using RELAP5 Code

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Cases of core coolant flow blockage transient have been simulated and analysed for the TRIGA IPR-R1 research reactor using the RELAP5-MOD3.3 code. The transients are related to partial and to total obstruction of the core coolant channels. The reactor behaviour after the loss of flow was analysed as well as the changes in the coolant and fuel temperatures. The behaviour of the thermal hydraulic parameters from the transient simulations was analysed. For a partial blockage, it was observed that the reactor reaches a new steady state operation with new values for the thermal hydraulic parameters. The total core blockage brings the reactor to an abnormal operation causing increase in core temperature.

1. Introduction

Operation of a research reactor is characterized by human actions and interventions on a daily basis, whether the reactor is at power operation or shutdown. Even with preventive measures, human actions or errors can still cause an initiating event [1]. The safety analysis of research reactors includes simulations of selected cases classified by the International Atomic Energy Agency, since the simulations are performed using nodalizations verified and validated by users of internationally recognized codes [1]. The thermal hydraulic analysis is considered as an essential aspect in the study of safety of nuclear power and research reactors, since it can predict proper working conditions, steady state and transient, thereby ensuring the safe operation of a nuclear reactor [2]. Among thermal hydraulic accidents are the loss of flow accident (LOFA) and loss of coolant accident (LOCA).

The aim is to investigate the behaviour of a TRIGA research reactor after partial and total coolant flow blockage in the thermal hydraulic channels (THC) characterizing a LOFA transient. The transients were simulated using a RELAP5 model. LOFA constitutes one of the most severe accidents that may occur during a research reactor lifetime [3]. The transient was simulated considering reactor power operation at 100 kW and 265 kW that are the main IPR-R1 power operation levels.

IPR-R1: General Description. The IPR-R1 is a reactor type TRIGA (Training, Research, Isotope, General Atomic), Mark-I model, manufactured by the General Atomic Company and installed at Nuclear Energy Development Centre (CDTN) of Brazilian Nuclear Energy Commission (CNEN), in Belo Horizonte, Brazil. It is a light water, graphite-reflected, open-pool type research reactor. Since 1970, IPR-R1 works at 100 kW but it is ready to operate at power of 250 kW. It presents low power and low pressure, for application in research, training, and radioisotopes production. The reactor is located in a 6.625 m deep pool with 1.92 m of internal diameter and filled with demineralized light water. A schematic reactor diagram is illustrated in Figure 1.

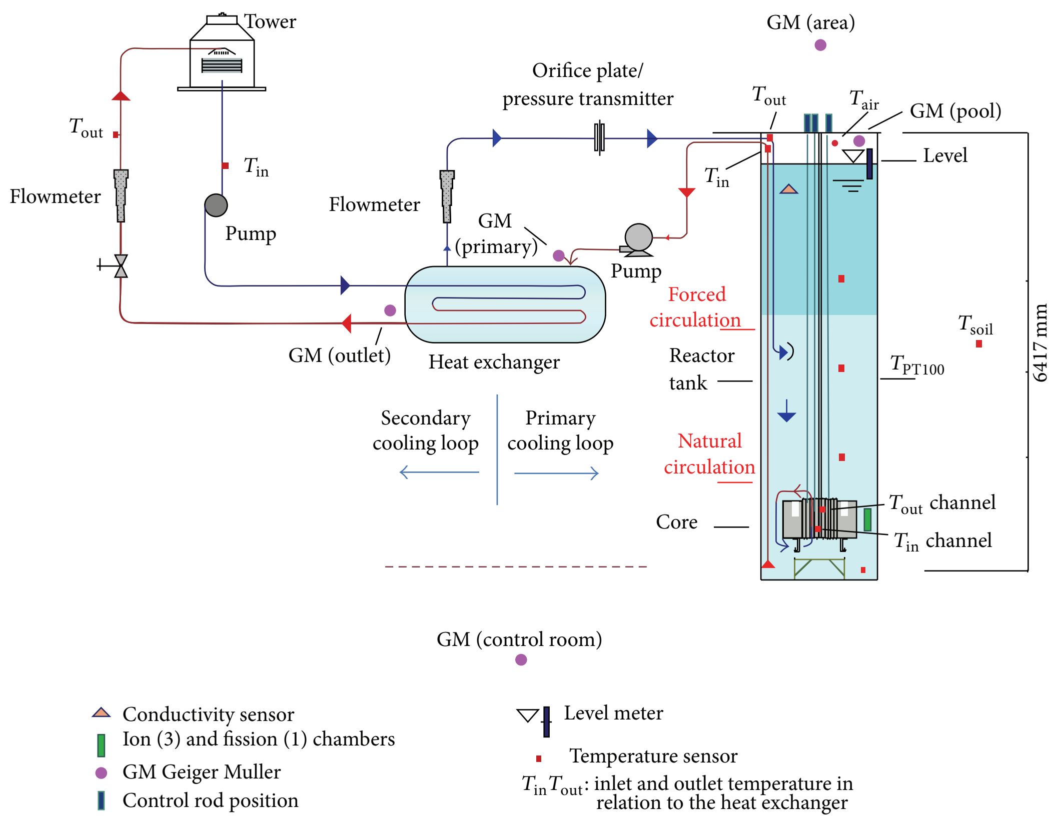


Figure 1: Schematic representation of the TRIGA IPR-R1 [17].

The water in the pool works as coolant, moderator, neutron reflector and it is able to assure an adequate biological radioactive shielding. The reactor cooling occurs predominantly by natural convection, with the circulation forces governed by the water density differences. The removal of the heat generated from the nuclear fissions is performed by pumping the pool water through a heat exchanger. The core has a radial cylindrical configuration with six concentric rings (A, B, C, D, E, and F) with 91 positions able to host either fuel rods or other components like control rods, reflectors, and irradiator channel. There are in the core 63 fuel elements constituted by a cylindrical metal cladding filled with a homogeneous mixture of zirconium hydride and uranium 20% enriched in ^{235}U isotope. These fuel elements have three axial sections, an upper and a lower reflector (graphite), and the central portion filled with fuel (U-ZrHx) [4]. The radial reflectors elements are covered with aluminum and filled with graphite having the same dimensions of the fuel elements.

The control rods are composed by boron carbide and aluminum cladding. In the center of the reactor, there is an aluminum tube (central thimble) for irradiation of experimental samples. This tube is removable and when it is not in use, the reactor pool water fills its volume. Furthermore, the core has an annular graphite reflector with aluminum cladding. Such annular reflector has a radial groove where a rotary rack is assembled for insertion of the samples to irradiation. In such rotary rack it is possible to place the samples in 40 different positions around the core. Moreover, tangent to annular reflector, there is a pneumatic tube where the samples also can be inserted for irradiation.

TRIGA reactors fuel has an important characteristic of intrinsic safety with large, prompt negative coefficient of reactivity that effectively controls the reactor in cases of positive reactivity insertions. In this way, any sudden reactivity addition causes an increase in power which heats the fuel-moderator material (U-ZrHx) instantaneously causing the number of fissions to immediately decrease because of neutron energy spectrum hardening within the fuel pin.

2. IPR-R1 Nodalization

2.1. Reactor Modelling. The radial power distribution, shown in Figure 2(a), was calculated in preceding works using the WIMSD4C and CITATION codes [5, 6] and also experimental data [7]. The radial factor is defined as the ratio of the average linear power density in the element to the average linear power density in the core.

RELAP5 nodalization for the IPR-R1, verified in previous work (with 91 THC) [8], was utilized in this simulation. The core model has 91 hydrodynamic channels (see Figure 2(b)). The complete reactor nodalization in the RELAP5 is presented in Figure 3 [8–10]. To simulate the cross flow among the THCs a total of 186 single junctions were used, interconnecting selected volumes at different high positions of the nodalized channels. These junctions have been added to the model to observe the behaviour of the coolant inside the core

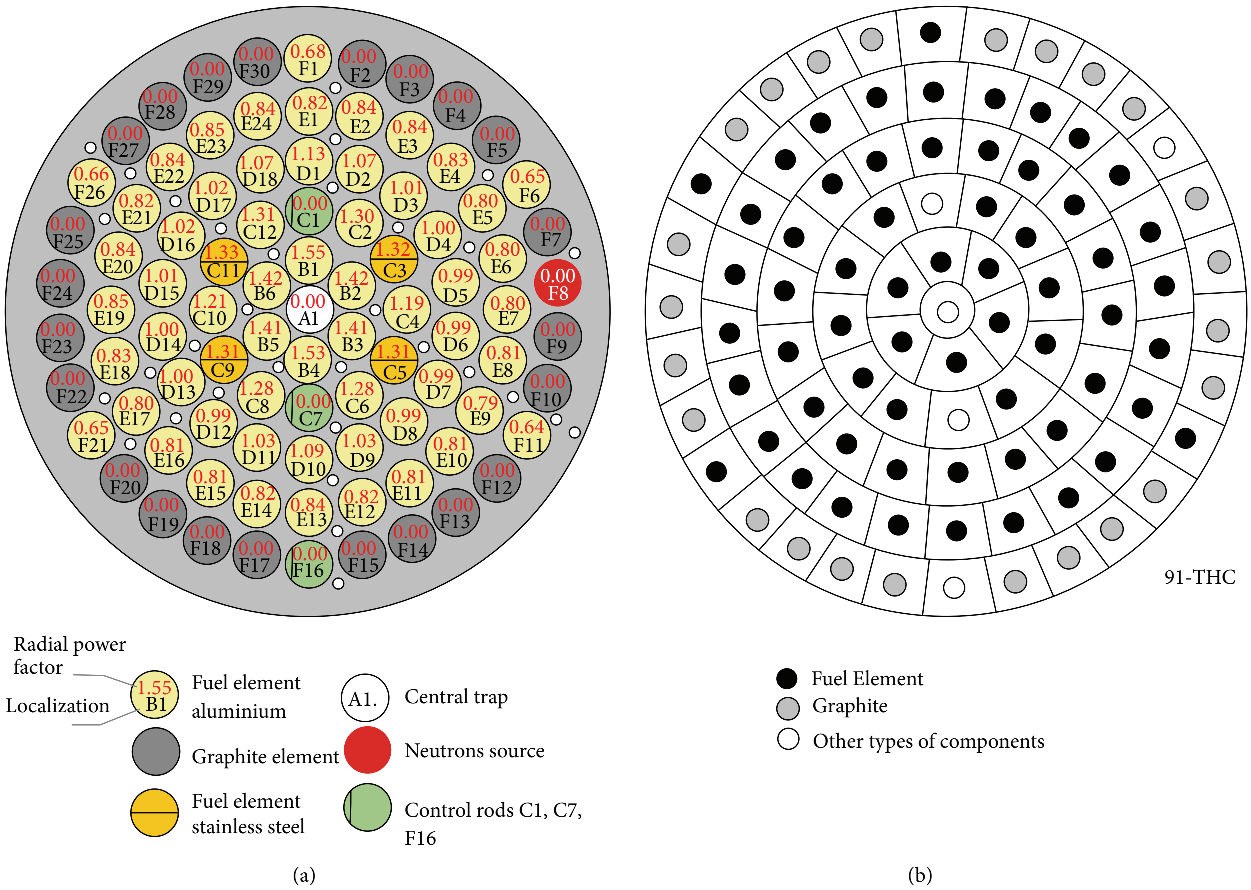


Figure 2: IPR-R1—radial relative power distribution (a); model of the 91 THC in the RELAP5 code (b).

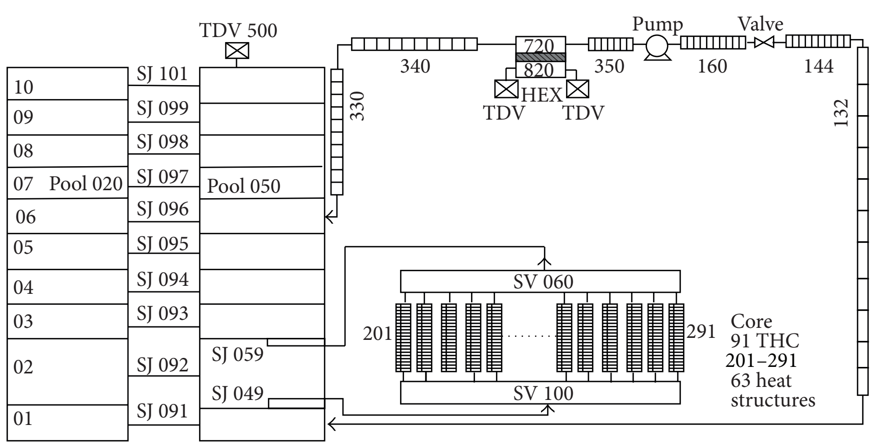


Figure 3: IPR-R1—Nodalization in the RELAP5-MOD3.3 code.

after the THC blockage transient. This nodalization is capable of simulating situations of forced coolant recirculation and also natural circulation.

2.2. Heat Structures Modelling. Each of the 63 fuel elements was modeled separately and 63 heat structure components, divided in 21 axial levels, were considered. The heat structures in RELAP5 permit the calculation of heat across the solid boundaries of the hydrodynamic volumes. Radially, the heat structures were divided in 17 mesh points considering the actual radius of the fuel (heat source), gap, and cladding. The fuel elements were modelled using cylindrical geometry and the characteristics of the materials were inserted to reproduce the heat transfer. For each material specified, corresponding thermal property data were entered to define the thermal conductivity and volumetric heat capacity as functions of temperature.

The axial power distribution was calculated considering a cosine profile, predicted using the data from Figure 2 and taking into account the power cutoff in the extremes of the element due the presence of the graphite. The point kinetics model was used in the current modeling, which assumes that changes in reactivity uniformly affect the core flux, resulting in no relative spatial variation over time. Point kinetics is not indicated when a pronounced flux spike and radial neutron wave occur.

3. THCs Blockage Transient

THCs blockage transient has been investigated using the RELAP5 code. For the open pool research reactors configuration, the probability of a blockage in the core upper plenum zone is higher than a blockage from the bottom [11]. This transient situation may be caused, for example, by swelling of the fuel or fall of some material in the reactor pool leading to blockage of one or more channels [12]. A fuel channel blockage event has different characteristics depending on the flow direction. Downward cooling flow can lead to blockage due to objects dropping into the pool. Upward cooling flow can lead to blockage due to objects inside the primary cooling system piping being dragged into the core by the action of the pump [1]. In this work, all the simulations considered the blockage at the core inlet.

Two cases of transient were considered. In the first case, the aim was to investigate a partial core blockage simulation; in the second one, it was considered a total core blockage. Both transient cases, described, respectively, in more details in Sections 4.1 and 4.2, were initiated after the calculation reached steady state condition. Two values of power operation, 100 kW and 265 kW, typical IPR-R1 power operation values, were considered. To perform the partial blockage simulation using the RELAP5, 37 components of type valves at the inlet of TH channels in the rings A, B, C, and D were used (Figure 4). The valves were closed at 4000 s. To simulate the total core blockage, the valve number 38 was closed at 4000 s, stopping totally the coolant flow in the core.

4. Simulation Results

Since each flow channel provides its own driving force, it is possible to consider flow channel independently in a steady state situation [13]. However, to perform this specific type of transient, the interaction between all the obstructed channels and all the adjacent channels must be considered. RELAP5 is a one-dimensional code and each hydrodynamic cell has two faces in the normal direction, at the inlet and outlet. Then, to more realistically analyse the core behaviour in the case of a channel blockage, it is necessary to consider the coolant cross flow. Therefore, the RELAP5 cross flow junction model [14] has been used to represent the multidimensional phenomena of the flow, interconnecting the channels using components of type single junctions.

4.1. Partial Core Coolant Flow Blockage. The transient begins at 4000 s. The coolant flow rate in the valves reaches the total obstructed condition about 5 s after the beginning of the event (Figure 5). Due the blockage at inlet of the THCs, the coolant flow changes direction and module to adequate the parameters for the new core operation conditions.

The mass flow rate at the inlet of the channel B1 decreases after the blockage event reaching zero value (Figure 6). The mass flow rate at the midheight of this channel increases and changes the flow direction. Due the THC obstruction, the coolant flow is redistributed among adjacent channels. After the transient, a new steady state condition is reached (Figure 7).

Figure 8(a) shows a detail of the nodalization where the junctions make possible the cross flow between THCs during the transient. The mass flow rate at the cross flow junctions between the THCs C2 and D3, in four axial levels, are represented in Figure 8(b). It is possible to observe that until 4000 s of calculation the mass flow rate in the junctions is practically zero. After the transient, the coolant flows also among the channels characterizing the cross flow and reproducing the actual situation.

After the beginning of the transient, the coolant temperature increases in the obstructed channels at midheight and, due to the change of flow direction, the outlet temperatures decrease. A new steady state is reached. As an example, Figure 9 shows the coolant temperature in three axial levels of the THC D4 at 100 kW and 265 kW of power operation.

After the transient, it is possible to observe void formation in the obstructed channels. Figure 10 shows the void fraction at the outlet of the blocked channels B2, C3, and D4.

4.1.1. Heat Transfer Analysis. The heat transfer from the cladding surface to the water occurs locally in the regime of subcooled nucleate boiling, so that the cladding surface temperature (T_{sur}) is the saturation temperature of the water plus the wall superheat, given by McAdams correlations [15]. Therefore, the cladding surface temperature, T_{sur}, can be found using the expression

$$T_{\text{sur}} = T_{\text{sat}} + \Delta T_{\text{sat}}, \tag{1}$$

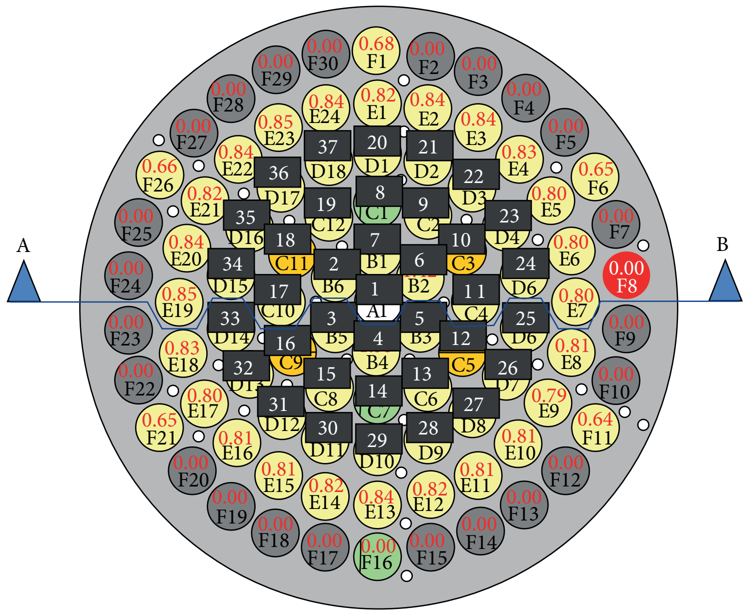
where T_{sat} is the water saturation temperature and it is equivalent to 111.37°C or 384.54°K at pressure of 1.5 bar [16]. The ΔT_{sat} is the wall superheat temperature that is given by

$$\Delta T_{\text{sat}} = 0.81 \left(q''\right)^{0.259}, \tag{2}$$

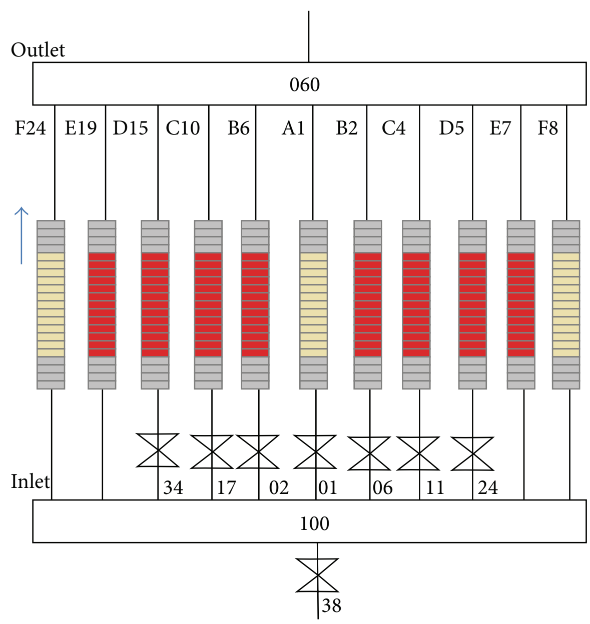
with temperature in °C and heat flux, q'', in W/m^2.

Specifically for the IPR-R1 TRIGA reactor conditions, the transition point between the single-phase convection regime to subcooled nucleate boiling regime is approximately at 60 kW of power operation [17].

Figure 11 shows the time evolution of the heat flux q'' and the heat transfer coefficient, h_s, in the heat structure B1 at



(a)



(b)

FIGURE 4: Core nodalization showing in details the positions of the valves used to perform the partial core blockage (a); central cross view of the core nodalization (b). The valve number 38, connected to the element number 100, is used to perform the total core blockage.

midheight and at 100 kW of power operation for RELAP5 calculation.

The values of q'' were taken from RELAP5 steady station calculation and ΔT_{sat} and T_{sur} were calculated using (1) and (2). The values for 100 kW and 265 kW are, respectively,

$$\Delta T_{sat(100\,kW)} = 0.81\,(72927.88)^{0.259} = 14.72^{\circ}\mathrm{C},$$
$$\Delta T_{sat(265\,kW)} = 0.81\,(193067.6)^{0.259} = 18.95^{\circ}\mathrm{C}. \quad (3)$$

Therefore

$$T_{(sur100\,kW)} = 126.09^{\circ}\mathrm{C},$$
$$T_{(sur265\,kW)} = 130.31^{\circ}\mathrm{C}. \quad (4)$$

These results means that the reactor regime is the subcooled nucleate boiling in which $T_{sur} > T_{sat}$, but $T_{fluid} < T_{sat}$.

Table 1 presents a comparison among experimental and calculated values of the fuel thermal parameters. The calculated parameters using RELAP5 code are taken at midheight

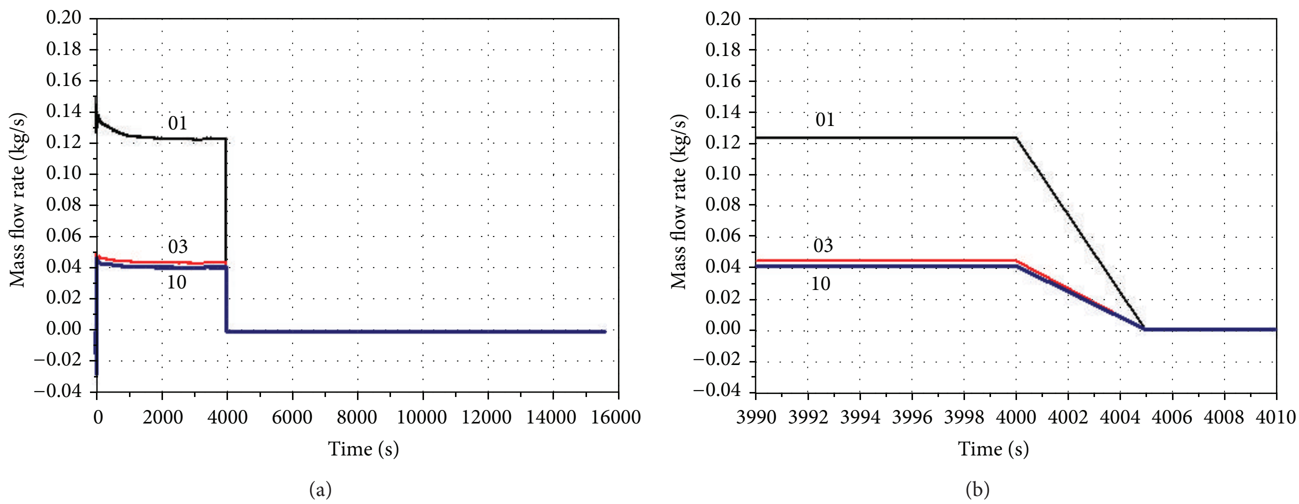


Figure 5: Mass flow rate evolution in valves 1, 3, and 10 at 265 kW of power operation tripped at 4000 s (a). The time window shows it in details (b).

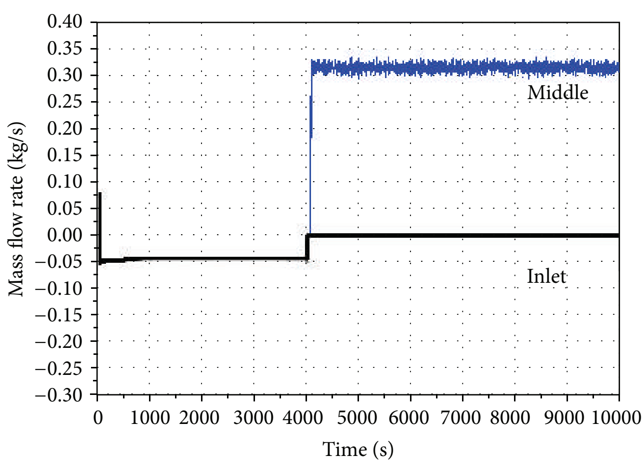


Figure 6: Inlet and midheight mass flow rate in the channel B1—before and after the blockage initiated at 4000 s (power of 265 kW).

(axial level 11) of B1 fuel element and are in good agreement with the experimental data [17].

The convective heat transfer coefficient, as it can be seen in Figure 11, increases after the blockage, reaching a new steady state value. As it was verified from the results obtained, the transient presented variation of the thermal hydraulic parameters that quickly reached new steady state values.

4.2. Total Core Blockage. In this situation all the thermal hydraulic channels were blocked using a component type valve (valve 38 in the Figure 4) closed at 4000 s of calculation, therefore, after the steady state conditions were reached. Figure 12 shows the mass flow rate at 100 and 265 kW of power operation for the THC D4. The mass flow rate, after the transient, oscillates around zero value.

After the beginning of the transient, the coolant temperature in the core increases reaching the saturation temperature. The regime now is the saturated or bulk boiling, with the coolant temperature remaining essentially constant and equal to T_{sat}. As an example, Figure 13 shows the coolant temperature in three axial levels of the THC B1 at 100 kW and 265 kW of power operation. The same behaviour is observed for all TH core channels.

Table 1: Experimental and calculated values of some fuel thermal parameters.

	Power (kW)	q'' (W/m^2)	ΔT_{sat} (°C)	Cladding temperature (°C)
Experimental	265.0	194613.0	19.00	130.4
RELAP5	265.0	193067.6	18.95	130.3

In addition, Figure 14 shows the void fraction and the coolant temperature time evolution in TH channel D4 at 265 kW.

Specifically to TRIGA reactors the limiting parameter is the fuel temperature. The maximum fuel temperature is related to the dehydrogenation of uranium and zirconium hydride and subsequent stress on the cladding. The temperature at which the dehydrogenation occurs in $UZrH_{1.0}$ is approximately 550°C. This temperature should not be exceeded to avoid volumetric expansion of the UZrH and deformation of the fuel element [4].

There is no significant increase in the fuel and cladding temperatures after the total blockage (Figure 15). Even with the loss of flow and the coolant reaching the saturation temperature, the fuel temperature remains below the safe threshold temperature.

The loss of flow accidents is part of a category that involves weak feedback effects when it occurs out of core. However, in the total core blockage, significant disturbance of the core was observed and also void formation was observed. Therefore, a more realistic analysis of this type of accident could be performed considering cross section variations. This is not possible to do using a point kinetics model.

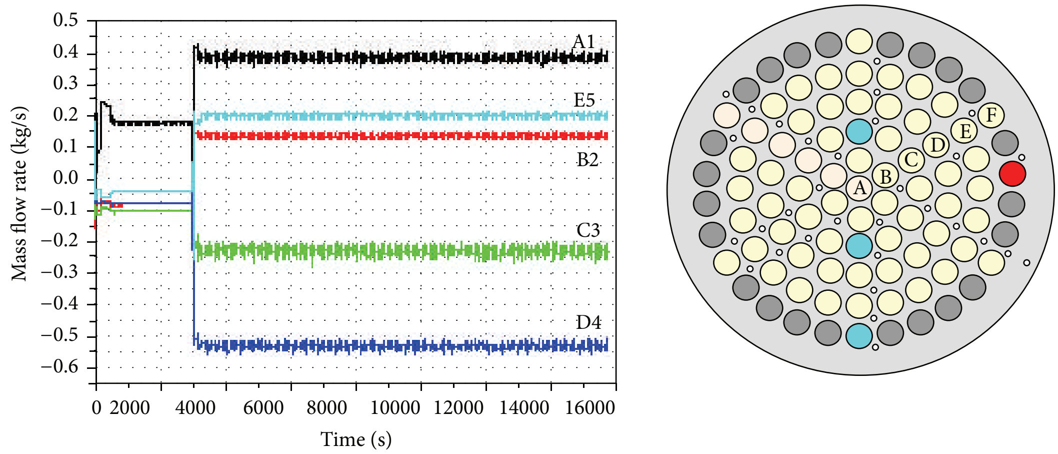


FIGURE 7: Outlet mass flow rate in the channels A1, B2, C3, D4, and E5 before and after the blockage (power of 265 kW).

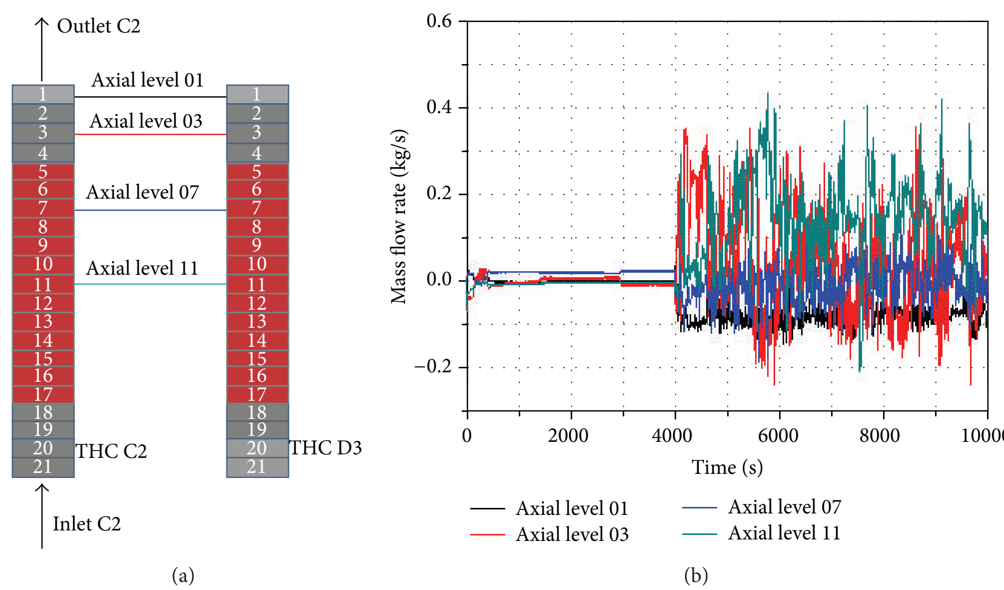


FIGURE 8: Details of the nodalization where the junctions make possible the cross flow between THCs during the transient (a). Mass flow rate at the cross flow junctions between the THCs C2 and D3, in four axial levels (b).

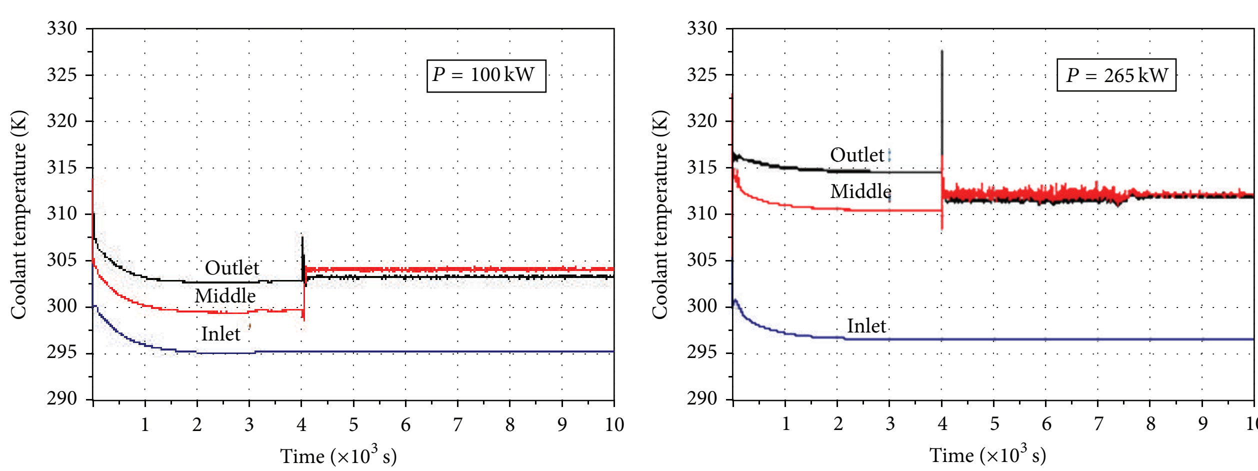


FIGURE 9: Coolant temperature at the THC D4 at 100 kW and 265 kW.

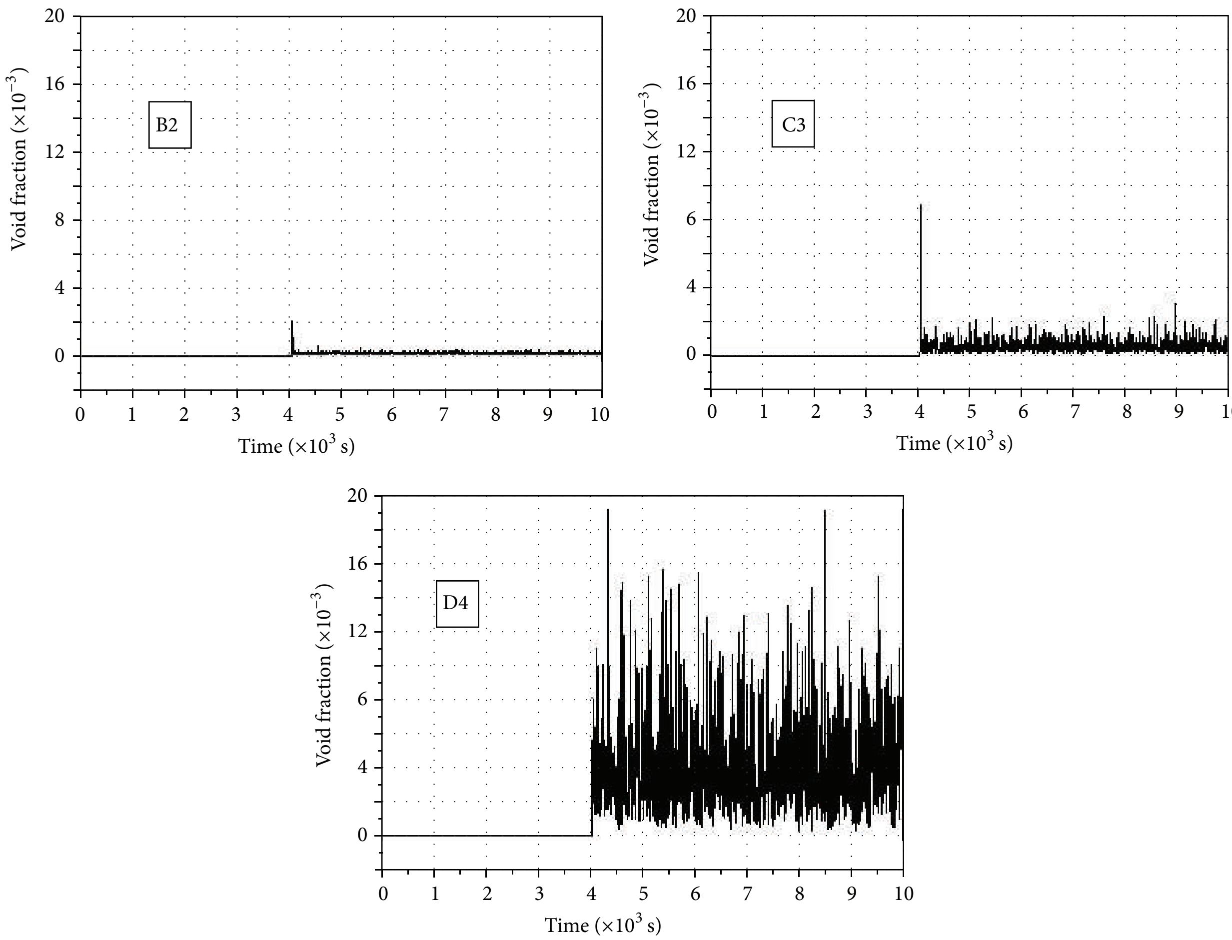


Figure 10: Void fraction time evolution at outlet of the thermal hydraulic channels B2, C3, and D4 after the transient at 100 kW.

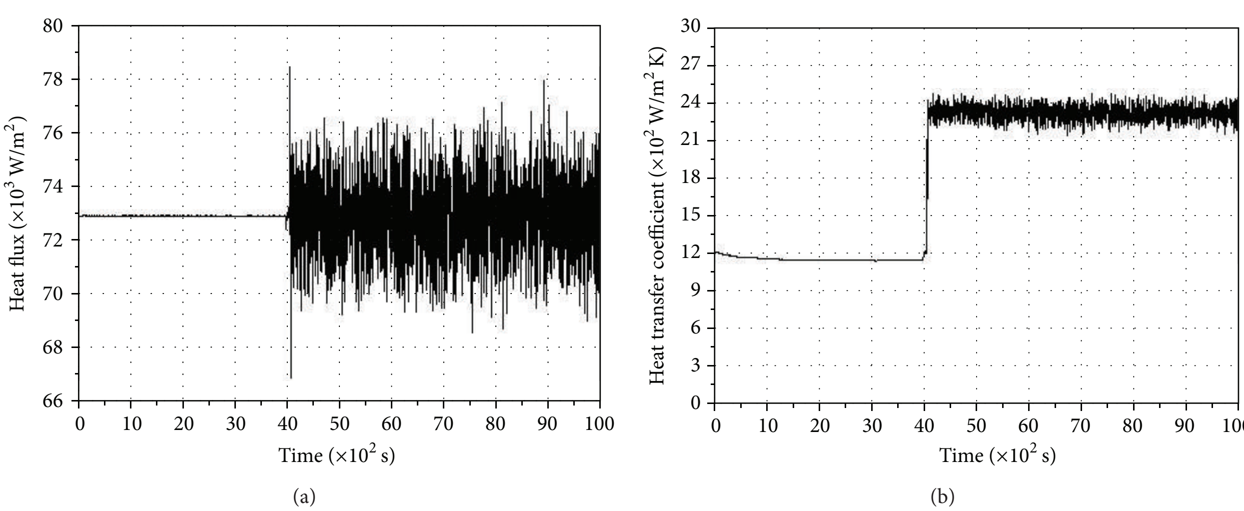
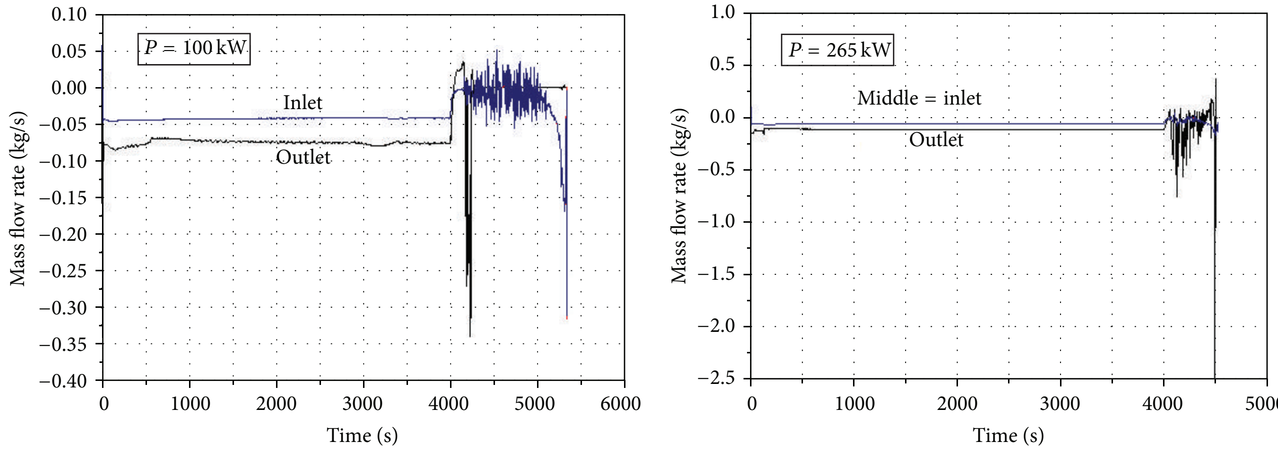


Figure 11: Heat flux (a) and heat transfer coefficient (b) to the heat structure associated with the THC B1, at 100 kW of power operation.

5. Conclusions

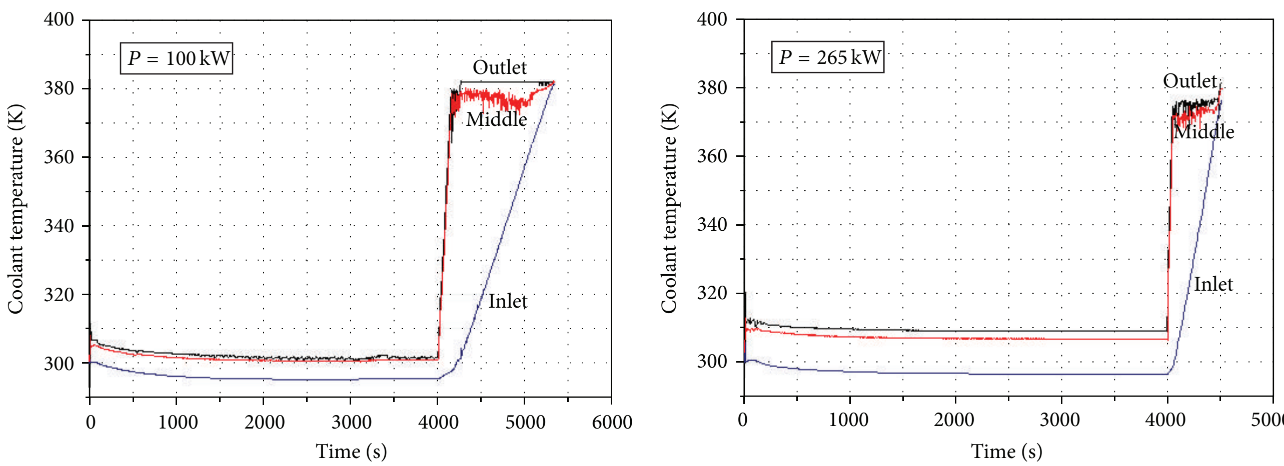
To perform the calculations a verified model in RELAP5/MOD3.3 was considered. The simulated transient characterizes a LOFA type accident. Total and partial core coolant blockage transient have been investigated for the IPR-R1 research reactor. In the simulation, the channels were blocked using components type valve in the inlet of the channels. In both cases, total and partial blockage, the cross flow modeling in the RELAP5 core nodalization is important in the analyses



(a)

(b)

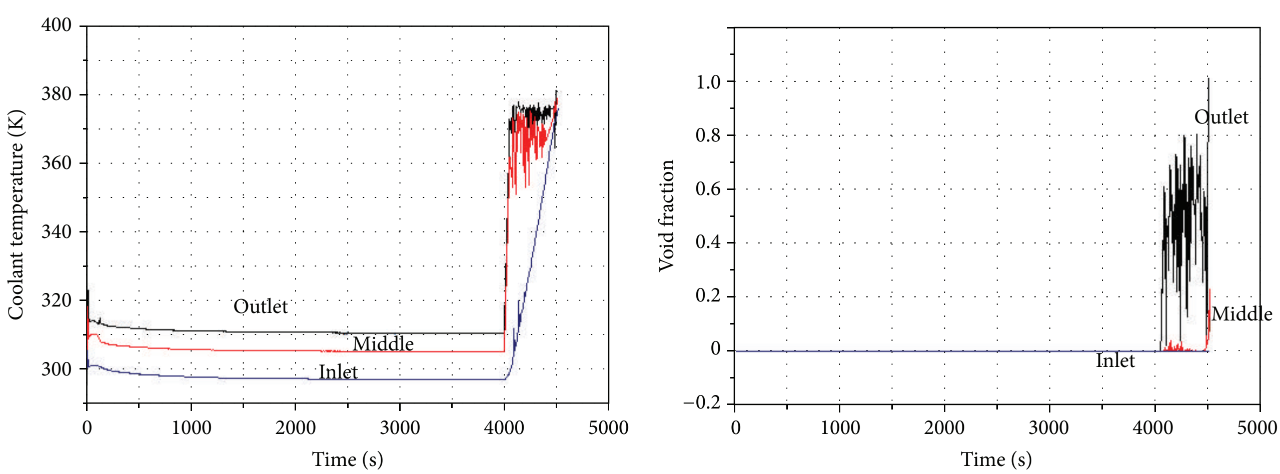
FIGURE 12: Mass flow rate at channel D4 after and before the total core blockage at 100 kW (a) and 265 kW (b) of power operation.



(a)

(b)

FIGURE 13: Coolant temperature in the channel B2 at 100 kW (a) and 265 kW (b) of power operation.



(a)

(b)

FIGURE 14: Coolant temperature (a) and void fraction (b) evolutions for the THC D4 at 100 kW.

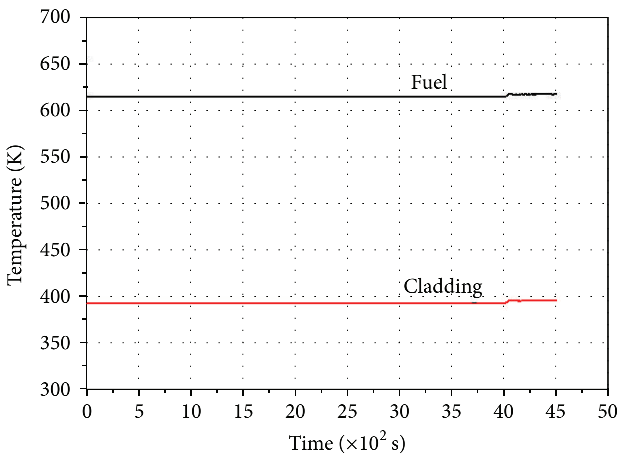


Figure 15: Fuel and cladding temperature evolutions for the B1 heat structure at 265 kW of power operation.

because it provides the coolant redistribution among adjacent channels after the blockage as it is the expected behaviour.

After the partial blockage transient, an increase in the coolant temperature of the blocked channels, change in the mass flow direction and modulus, and void formation were observed. In spite of this, the reactor presented safe behaviour after the transient reaching a new steady state. However, the model is not capable of reproducing exactly the behaviour of the core due to the changes in the macroscopic cross sections that are directly affected by coolant density and temperature.

In the case of the total blockage, the saturation coolant temperature is reached into the core, and the coolant boils in few minutes after the beginning of the transient. One more time, in an actual situation, the intrinsic safety characteristic of fuel should work to control the reactor. It was observed that a total core blockage disturbs significantly the core and the reactivity feedback effects should be considered. A neutronic analysis, in this case, could be necessary to investigate how the neutron flux is perturbed in the core after the total blockage transient.

Conflict of Interests

The authors declare that there is no conflict of interests regarding the publication of this paper.

Acknowledgments

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