ANALYSIS OF HYPOTHETICAL CORE BLOCKAGE CASES IN A RESEARCH REACTOR USING THE THERMAL-HYDRAULIC CODE RELAP5

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ABSTRACT

A RELAP5 model for simulation of the IPR-R1 TRIGA research reactor was developed and validated for steady-state and transient situations. The RELAP5 model validated for the IPR-R1 TRIGA reproduces the actual steady-state reactor behavior in good agreement with the available data, as well as, a loss of flow transient. In the present work, studies of cases related with core blockage transient have been investigated during normal operation of the IPR-R1 using the RELAP5 code. The considered transients are related to partial and also to total obstruction of coolant core channels. The reactor behavior after the loss of flow was analyzed as well as the changes in the coolant and fuel temperatures. The results of the thermal-hydraulic parameters from the transient simulations presented behavior as expected. For a partial blockage, it was observed that the reactor reaches a new steady-state operation with new values for the thermal-hydraulic parameters. However, the total core blockage brings the reactor to dangerous operation according with the increase in core temperature observed and presented in this work.

1. INTRODUCTION

Operation of a research reactor is characterized by human actions and interventions on a daily basis, whether the reactor is at power or shutdown. Even with all preventive measures in place, human actions or errors can still cause an initiating event [15]. The safety analysis of research reactors includes simulations of selected cases classified by the International Atomic Energy Agency, since the simulations are performed using validated nodalizations and internationally recognized, accepted and validated best estimate codes. The thermal-hydraulic analysis is considered as an essential aspect in the study of safety of nuclear reactors, since it can predict proper working conditions, steady-state and transient, thereby ensuring the safe operation of a nuclear reactor [1]. Among thermal-hydraulic accidents are loss of flow accident (LOFA) and loss of coolant accident (LOCA).

In the last decades, several codes have been developed to predict the thermal-hydraulic behaviour of nuclear reactors — like ATHLET, CATHARE, RELAP, RETRAN. However, these codes were developed to power reactors perform. To extend the application for the analyses of research reactor some modifications or addition of some procedures have been done, as described for example in [2].
The aim of this work is to investigate the behavior of TRIGA nuclear research reactor after a partial and total blockage of its thermal-hydraulic (TH) channels characterizing a LOFA transient by simulating the reactor with a RELAP5 model. LOFA constitutes one of the most severe accidents that may occur during a research reactor lifetime [12]. The transient was simulated considering power operation at 100 kW and 265 kW.

1.1. 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 moderated and cooled, 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, 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 Fig. 1.

The water in the pool has function of cooling, as well as 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 pumping the pool water through a heat exchanger. The core has a radial cylindrical configuration with six concentric rings (A, B, C, D, E, F) with 91 channels able to host either fuel rods or other components like control rods, reflectors and irradiator channels. 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 lower reflector (graphite), and the central portion filled with fuel (U-ZrHx) [9]. The point kinetics model was used in the current model.

The radial power distribution (Fig. 2a) was calculated in preceding works using the WIMSD4C and CITATION codes [3, 4] and also experimental data [5]. 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.
2. NODALIZATION

In this work was utilized a RELAP5 nodalization validated in previous work (91-THC) [8]. In such nodalization each one of the 63 fuel elements, the regulation and control rods, the reflector elements and the neutron source were modeled separately and were associated with 91 corresponding hydrodynamic pipe components constituting 91 hydrodynamic channels as it can be verified in the Fig. 2b. The Fig. 3 presents the total nodalization of the IPR-R1 in the RELAP5 [6, 7, 8].

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3. TH CHANNEL BLOCKAGE TRANSIENT

A transient event, characterized by TH channels blockage 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 [13]. This transient situation may be caused by swelling of the fuel, fall of some material in the reactor pool, leading to blockage of one or more channels [2]. According with IAEA [15], 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.

In this work, two cases of transient were considered. In the first case the aim was to investigate a partial core blockage, and in the second one it was considered a total core blockage. Both transient cases began after the calculation reached a steady-state condition. Two power operations, at 100 and 265 kW, were also considered. To perform the partial blockage simulation using the RELAP5, it was used component type valve at the inlet of thermal-hydraulic channels in the rings A, B, C and D. Fig. 4 shows (cross view) a small part of the core nodalization with valves from 1 up to 7. In the total, there were inserted 37 valves to perform the blockage. These valves were closed at 4000 s of calculation after the system to reach steady-state condition (valve 8 remained open). To simulate the total core blockage, the valve 8 was closed at 4000 s avoiding the coolant flow for all thermal-hydraulic channels.
4. **CALCULATION AND RESULTS**

Since each flow channel provides its own driving force, it is possible to consider flow channel independently [11]. However, to perform this 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. This situation doesn’t represent the actual behavior mainly when in the case of a channel blockage. Therefore, the RELAP5 crossflow junction model [10] has been used to represent the multi-dimensional 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 of calculation. As it can be seen in the Fig. 5, the coolant flow rate in the valves reaches the total obstructed condition about 5 s after the beginning of the event. Due the blockage at inlet of thermal-hydraulic channels, the coolant flow changes direction and module to adequate for the new situation.

In the Fig. 6, it is possible to see that the mass flow rate at the inlet of the channel B1 decreases after blockage reaching zero value. The mass flow rate at the mid-height of the same channel increases and changes the direction. Due to channels obstruction, the coolant flow is redistributed among adjacent channels. After the transient, a new steady-state condition is reached, as it is shown clearly in Fig. 7.
Figure 5. Mass flow rate evolution in some valves at 250 kW of power operation tripped at 4000 s (left). The time window shows it in details (right).

Figure 6. Inlet and mid-height mass flow rate in the channel B₁ – before and after the blockage.

After the beginning of the transient, the coolant temperature increases in the obstructed channels at mid-height and, due to the change of flow direction, the outlet temperatures presents a decrease. A new steady-state is reached very speedily. As an example, Fig. 8 shows the coolant temperature in three axial levels of the TH channel D₁ at 100 kW and 265 kW of power operation.
Figure 7. Outlet mass flow rate in the channels A_1, B_2, C_3, D_4 and E_5 before and after the blockage.

Figure 8. Coolant temperature at the TH channel D_4 at 100 kW and 265 kW.
After the transient, it is possible to observe void formation in the obstructed channels. Fig. 9 shows the void fraction at the outlet of some blocked channels.

![Figure 9](image)

**Figure 9. Void fraction evolution at outlet of the thermal-hydraulic channels B₂, C₃ and D₄ after the transient at 100 kW.**

The heat transfer from the cladding surface to the water occur locally in the regime of subcooled nucleate boiling, so that the cladding surface temperature ($T_{\text{sur}}$) is the saturation temperature of the water plus the wall superheat, given by McAdams correlations [17]. Therefore, the cladding surface temperature, $T_{\text{sur}}$, can be found using the expression:

$$T_{\text{sur}} = T_{\text{sat}} + \Delta T_{\text{sat}},$$  \hspace{1cm} (1)

where $T_{\text{sat}}$ is the water saturation temperature, This quantity is equivalent to 111.37 °C or 384.54 K at pressure of 1.5 bar [16]. The $\Delta T_{\text{sat}}$ is the wall superheat which is given by:

$$\Delta T_{\text{sat}} = 0.81 \left( q'' \right)^{0.259},$$  \hspace{1cm} (2)

with temperature in °C and heat flux in W/m².

Specifically for the IPR-R1 TRIGA nuclear reactor conditions, the transition point between the single-phase convection regime to subcooled nucleate boiling regime is approximately at 60 kW of power operation [14].

Fig. 10 shows the time evolution of the heat flux $q''$ and the heat transfer coefficient, $h_s$, in the heat structure $B_1$ at mid-height and at 100 kW of power operation.
The values of $q''$ were taken from RELAP5 steady station calculation and $\Delta T_{\text{sat}}$ and $T_{\text{sur}}$ were calculated using the equations 1 and 2. The values for 100 kW and 265 kW are respectively:

$$\Delta T_{\text{sat}(100 \, \text{kW})} = 0.81(72927.88)^{0.259} = 14.722259 \degree C$$

$$\Delta T_{\text{sat}(265 \, \text{kW})} = 0.81(193067.6)^{0.259} = 18.944526 \degree C$$

Therefore:

$$T_{\text{sur}(100 \, \text{kW})} = 126.092 \degree C$$

$$T_{\text{sur}(265 \, \text{kW})} = 130.314 \degree C$$

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

Table 1 presents experimental [14] and calculated values of the fuel thermal parameters. The calculated parameters using RELAP5 code are taken at mid-height (axial level 11) of B₁ fuel element and are in very good agreement with the experimental data.

<table>
<thead>
<tr>
<th>Power (kW)</th>
<th>$q''$ (W/m²)</th>
<th>$\Delta T_{\text{sat}}$ (°C)</th>
<th>Cladding Temperature (°C)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Experimental</td>
<td>265.0</td>
<td>194613.0</td>
<td>19.00</td>
</tr>
<tr>
<td>RELAP5</td>
<td>265.0</td>
<td>193067.6</td>
<td>18.95</td>
</tr>
</tbody>
</table>

The convective heat transfer coefficient, as it can be seen in the Fig. 10, increases after the blockage, reaching a new stationary value.

Loss of flow accidents are part of a category that involving weak feedback effects and it is mainly related to the thermal-hydraulic events during the failures of the core cooling system.
In fact, 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 8 in the Fig. 4) closed at 4000 s of calculation after the steady-state conditions reached. Fig. 11 shows the mass flow rate at 100 and 265 kW of power operation for the thermal-hydraulic channel D4. The mass flow rate, after the transient, oscillates around zero.

![Figure 11. Mass flow rate at channel D4 after and before the total core blockage at a) 100 kW and b) 265 kW of power operation.](image-url)

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 $T_{\text{sat}}$. As an example, Fig. 12 shows the coolant temperature in three axial levels of the TH channel B1 at 100 kW and 265 kW of power operation. The same behavior is observed for all TH core channels.

In addition, Fig. 13 shows the void fraction and the coolant temperature 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 de-hydrogenation of uranium and zirconium hydride and subsequent stress on the cladding. The temperature at which the de-hydrogenation 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.

As we can see in Fig. 14, the fuel and cladding temperatures increase a few after the total blockage. Although of the loss of flow and despite the coolant has reached the saturation temperature, the fuel temperature remains below the safe threshold temperature.
As it was explained before, the loss of flow accidents are part of a category that involves weak feedback effects being mainly related to the thermal-hydraulics events. However, as it was observed from the results of the total core blockage simulation, such transient disturbs significantly the core and it can lead to considerable damage. Therefore, a more realistic analysis of this type of accident could be performed using a coupled neutronic and thermal-hydraulic methodology. This will be the next step of this work.

5. CONCLUSIONS

In this work, total and partial core coolant blockage transient have been investigated considering two reactor power operations (100 and 250 kW) in the IPR-R1 research reactor. To perform the calculations a validated model in RELAP5/MOD3.3 was considered. The simulated transient characterizes a LOFA type transient. In the simulation, the channels were blocked using components type valve in the inlet of the channels. After the partial blockage transient, it was observed an increase in the coolant temperature of the blocked channels, change in the flux direction and void formation. In spite of this, the reactor presented a safe behaviour after the transient reaching a new stead state. However, for the total blockage case, the saturation coolant temperature is reached into the core, and the coolant boils in few minutes after the beginning of the transient bringing the reactor to a dangerous operation. The cross flow modeling in the RELAP5 core nodalization is very important in this type of analyses because it provides the coolant redistribution after the channels blockage approximating more from an actual situation.

It was observed that a total core blockage disturbs significantly the core making possible the appearing of reactivity feedback effects. In this way, the next step of this work is to use a coupled neutronic and thermal-hydraulic methodology to obtain a more realistic analysis for this type of accident.
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