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# Thermal hydraulic analysis of the IPR-R1 TRIGA research reactor using a RELAP5 model

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ABSTRACT

The RELAP5 code is widely used for thermal hydraulic studies of commercial nuclear power plants. Current investigations and code adaptations have demonstrated that the RELAP5 code can be also applied for thermal hydraulic analysis of nuclear research reactors with good predictions. Therefore, as a contribution to the assessment of RELAP5/MOD3.3 for research reactors analysis, this work presents steady-state and transient calculation results performed using a RELAP5 model to simulate the IPR-R1 TRIGA research reactor at 50 kilowatts (kW) of power operation. The reactor is located in the Nuclear Technology Development Center (CDTN), Brazil. It is a 250 kW, light water moderated and cooled, graphite-reflected, open pool type research reactor. The development and the assessment of a RELAP5 model for the IPR-R1 TRIGA are presented. Experimental data were considered in the process of the RELAP5 model validation. The RELAP5 results were also compared with calculated data from the STHIRP-1 (Research Reactors Thermal Hydraulic Simulation) code. The results obtained have shown that the RELAP5 model for the IPR-R1 TRIGA reproduces the actual steady-state reactor behavior in good agreement with the available data.


## 1. Introduction

In general, the purpose of nuclear research reactors is not for energy generation reaching maximum power of about 100 megawatts (MW). They are commonly applied for neutrons generation for several different purposes. However, specific features are necessary to ensure safe utilization of these installations and special attention should be focused for their safety aspects. Therefore, the current enlarged commercial exploitation of nuclear research reactors has increased the consideration to their corresponding safety issues (Adorni, 2007; Bokhari et al., 2002; Khater et al., 2007; Khedr et al., 2005).

Several codes (as, for example, the RELAP5) have been used for system safety analysis and valuation of specific perturbation plant processes. RELAP5 computer code is a LWR transient analysis code developed mainly by the Idaho National Engineering Laboratory (INEL) for the U.S. Nuclear Regulatory Commission (NRC) for use in rulemaking, licensing audit calculations, evaluation of operator guidelines, and as a basis for a nuclear plant analyzer (US NRC, 2001). The RELAP5 system code was developed to simulate transient scenarios in power reactors such as PWR and BWR. However, some recent works as, for example (Antariksawan et al., 2005; Khedr et al., 2005; Marcum et al., 2010), have been performed to investigate the applicability of the code to research reactors operating conditions (TRIGA 2000, MTR, Oregon State TRIGA, respectively).

The IPR-R1 TRIGA Mark-I model is installed at the Nuclear Technology Development Center (CDTN) of Brazilian Nuclear Energy Commission (CNEN), in Belo Horizonte City, Brazil, and it is in operation since 1960. The IPR-R1 has been modeled using the RELAP5 code with the aim to reproduce the measured steady-state conditions. The version MOD3.3 was used to perform the simulations. The validation has been verified against experimental data (at 50 kW of power). The RELAP5 results were also compared with data obtained using the STHIRP-1 code (Veloso, 2004) developed at the Nuclear Engineering Department in the Federal University of Minas

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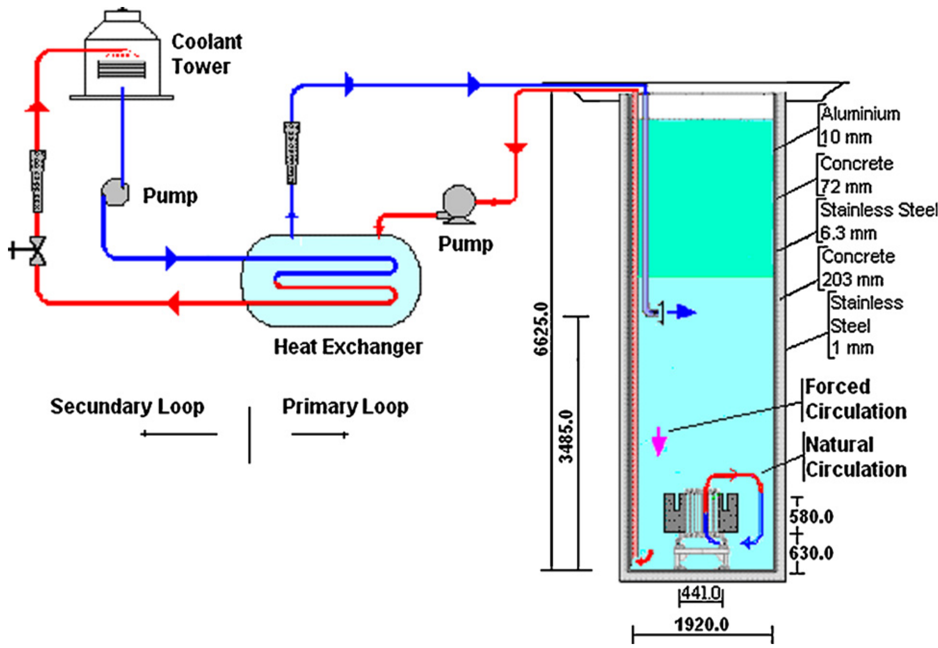


**Fig. 1.** Schematic representation of the IPR-R1 (out of scale, measure in mm).

Gerais, Brazil, to simulate the IPR-R1 reactor. The STHIRP-1 computer program uses the principles of the sub-channels analysis and it simulates, under steady-state and transient situations, the thermal and hydraulic phenomena occurring inside the core of water cooled research reactor under natural convection conditions. The models and empirical correlations necessary to describe the flow phenomena which cannot be described by theoretical relations were selected according to the characteristics of the IPR-R1 TRIGA operation. The computer program was validated against the IPR-R1 TRIGA model (Veloso, 2004). The comparison between STHIRP-1 results and the experimental data indicate that the code reproduces the steady-state reactor behavior with good agreements.

## 2. Reactor description

The TRIGA (Training, Research, Isotope, General Atomic) research reactors are constructed in a variety of configurations and capabilities, with steady-state power levels ranging from 20 kW up to 16 MW offering true “inherent safety”. The IPR-R1 is a reactor type TRIGA. It presents low power (250 kW), low pressure, for application in research, training and radioisotopes production. The reactor is housed 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 acts mainly as cooling, as well as moderator, neutron reflector and biological shielding for the core radiation. The reactor cooling occurs predominantly by natural convection, with the circulation forces governed by the water density differences. The heat removal generated from the nuclear fissions is performed pumping the pool water through a heat exchanger. The core is constituted by a 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 (see Fig. 2). There are 63 fuel elements constituted by a cylindrical metal cladding filled with the fuel, being 59 elements covered with aluminum and 4 elements with stainless steel.

The fuel material is a homogeneous alloy composed by zirconium hydride and uranium enriched at 20% in the $^{235}$U isotope. The radial power distribution was calculated in preceding works using the WIMSD4C and CITATION codes (Dalle, 2003; Dalle et al., 2002) and also experimental data (Veloso, 2004; Souza, 1999). 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. Fig. 2 shows the radial relative power distribution. The main general IPR-R1 TRIGA features are shown in Table 1 and the specific fuel element features are presented in Table 2 considering both aluminum and stainless steel (SS) cladding (CDTN/CNEN, 2000).

It is possible to include, in the RELAP5 model, reactivity (or scram) curves from the general tables or control variables that

**Table 1**
General reactor features.

| | |
|---|---|
| Core power (kW) | 250.0 |
| Delayed neutron fraction | 0.0079 |
| Prompt neutron generation time (s) | $10^{-4}$ |
| Isothermal coefficient (cents/kW) | 0.44 |
| Temperature reactivity feedback (cents/°C) | −1.1 |
| Pressure of operation (kPa) | 158.7 |
| Main moderator | Zirconium hydride (hydrogen) |
| Cladding | Aluminum or stainless steel (SS) |
| Coolant | Light water |
| Reflector | Graphite |

**Table 2**
Fuel element features.

| | Al cladding | SS cladding |
|---|---|---|
| Number of fuel elements | 59 | 4 |
| Fuel | $U\text{-}ZrH_{1.0}$ | $U\text{-}ZrH_{1.6}$ |
| Zr concentration (wt.%) | 91.0 | 89.9 |
| U concentration (wt.%) | 8.0 | 8.5 |
| H concentration (wt.%) | 1.0 | 1.6 |
| $^{235}$U enrichment | 20% | 20% |
| Cladding material | Al 1100-F | SS AISI-304 |
| Fuel diameter (mm) | 35.6 | 36.3 |
| Total height (mm) | 722.4 | 720.6 |
| Gap width (mm) | 0.09 | 0.14 |
| Total diameter (mm) | 37.3 | 37.6 |
| Active height (mm) | 355.6 | 381.0 |
| Graphite reflector height (mm) | 101.6 | 88.1 |
| Surface of heat transfer ($m^2$) | 2.46 | 0.18 |
| Gap Material | Helium | Helium |

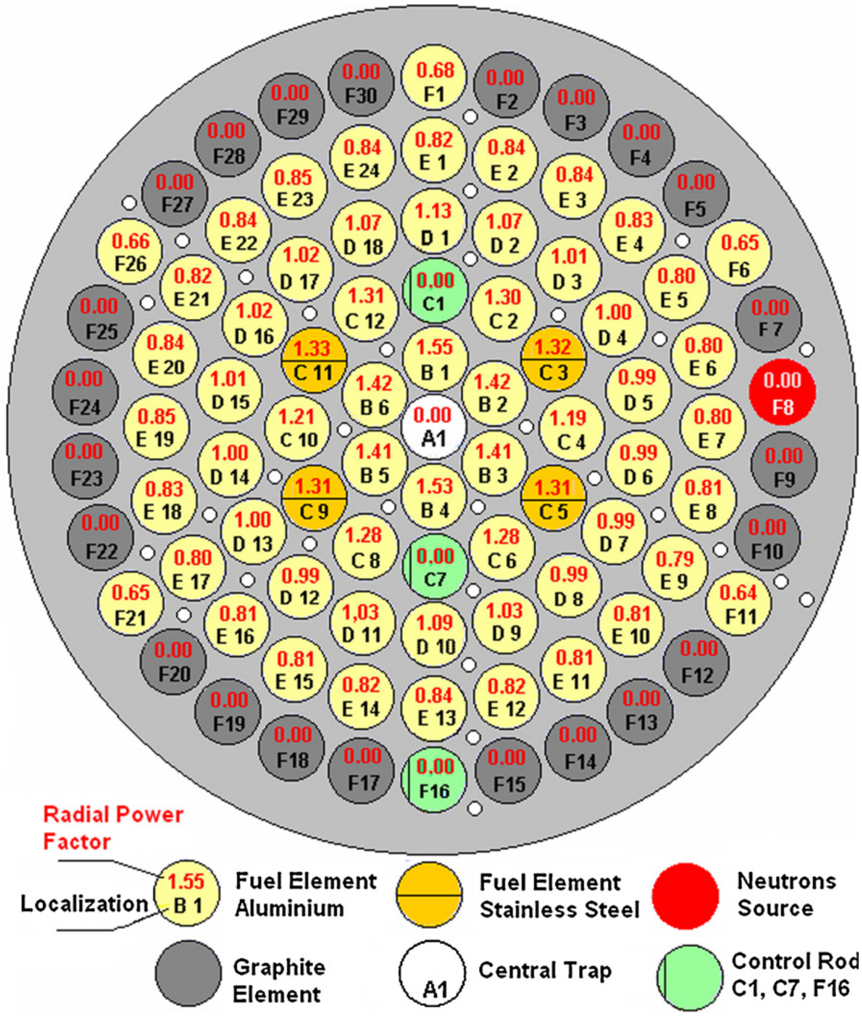


**Fig. 2.** Radial relative power distribution.

contribute to reactivity feedback described in specific cards. For example, in the "density reactivity table", one or more pairs of numbers are entered to define reactivity as a function of moderator density. Moreover, using the "Doppler reactivity table", one or more pairs of numbers are entered to define Doppler reactivity as a function of averaged fuel temperature. For the present model, a Doppler reactivity table with data from experimental measurements (Mesquita and Souza, 2010) has been used in the simulations.

The prompt temperature coefficient of reactivity is a very important safety parameter of research reactors. Specifically for the TRIGA reactors, the main moderator is the hydrogen that is mixed with the fuel itself. If the fuel temperature increases, the neutrons inside the hydrogen-containing fuel rod become warmer than the neutrons outside in the cold water, bringing to less fission in the fuel and escape into the surrounding water. Consequently, the reactor automatically reduces the power within a few thousandths of a second, faster than any engineered device can operate. The inherent safety of the IPR-R1 reactor arises from the prompt negative temperature reactivity coefficient $(-1.1 \pm 0.2)$ cents/$^{\circ}$C which effectively limits the power when excess reactivity is suddenly inserted (Mesquita and Souza, 2010). This characteristic of the fuel elements, that gives a high negative prompt temperature coefficient, is the main reason of the inherent safety behavior of the TRIGA reactors.

## 3. Thermal hydraulic model and calculation

Aiming to simulate the IPR-R1 TRIGA reactor using the RELAP5/MOD3.3 code, the reactor pool was firstly modeled using a pipe component composed by ten volumes. A time dependent volume was used to simulate the atmospheric pressure on the pool surface. Each of the 63 fuel elements was modeled separately and 63 heat structure (HS) components were associated with 13 corresponding hydrodynamic pipe components constituting 13 hydrodynamic channels (201–213), as can be verified in Fig. 3. Table 3 presents some characteristics of the 13 thermal-hydraulic (TH) regions.

The natural convection system and the primary loop circulation have been modeled. The secondary loop, composed mainly by the external cooling tower was not modeled in the present nodalization because the primary circuit was sufficient to guaranty the heat removal of the coolant. Fig. 4 shows the developed layout nodalization of the IPR-R1 TRIGA for the RELAP5 code.

The point kinetics model was used in the current model. A detailed representation of each element is, however, essential to properly take into account the radial power distribution associated with the position of the fuel elements. The axial power distribution was calculated considering a cosine profile and taking into account the power cut off in the extremes of the element due the presence of the graphite as it is sketched in Fig. 5. This modeling procedure

**Table 3**
General characteristics of the 13 TH regions.

| TH channel | Number of fuel elements | TH channel identifier | Mass flow area ($m^2$) | HS identifier | HS position (see Fig. 2) |
|---|---|---|---|---|---|
| 1 | 6 | 201 | 0.003595 | 201–206 | B1, B2, B3, B4, B5, B6 |
| 2 | 5 | 202 | 0.002779 | 207–211 | C2, D2, C3, D3, D4 |
| 3 | 5 | 203 | 0.002779 | 212–216 | C4, C5, D5, D6, D7 |
| 4 | 4 | 204 | 0.002759 | 217–220 | C6, D8, D9, D10, |
| 5 | 5 | 205 | 0.002779 | 221–225 | C8, C9, D11, D12, D13 |
| 6 | 5 | 206 | 0.002779 | 226–230 | C10, C11, D14, 15, D16 |
| 7 | 4 | 207 | 0.002756 | 231–234 | C12, D17, D18, D1 |
| 8 | 5 | 208 | 0.005477 | 235–239 | E2, E3, E4, E5, F6 |
| 9 | 5 | 209 | 0.005495 | 240–244 | E6, E7, E8, E9, F11 |
| 10 | 4 | 210 | 0.005436 | 245–248 | E10, E11, E12, E13 |
| 11 | 5 | 211 | 0.005477 | 249–253 | E14, E15, E16, E17, F21 |
| 12 | 5 | 212 | 0.005477 | 254–258 | E18, E19, E20, E21, F26 |
| 13 | 5 | 213 | 0.005477 | 259–263 | E22, E23, E24, E1, F1 |
| | Total = 63 | | | | Total = 63 |

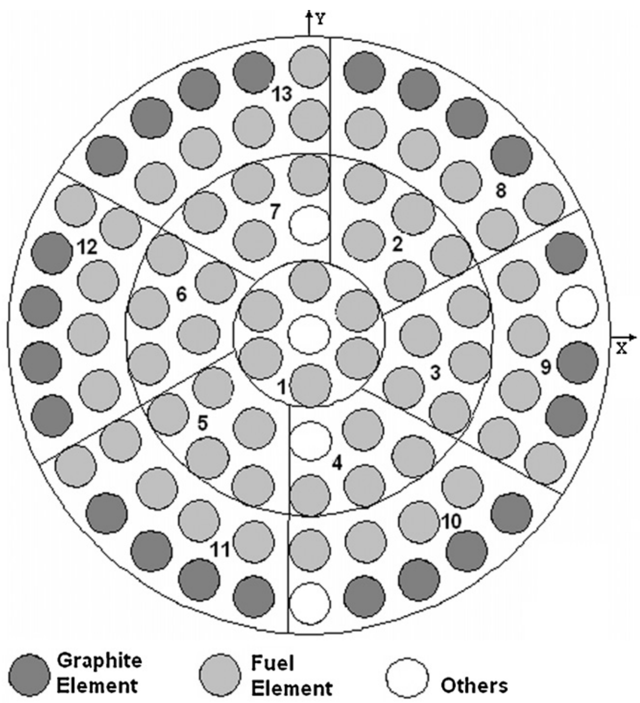


**Fig. 3.** Planar core representation showing the model with 13 TH regions.

is approximated but it is sufficient to maintain the actual axial and radial power distribution fixed.

### 3.1. Steady-state calculation

To validate a RELAP5 nodalization it must to demonstrate that the model reproduces the measured steady-state conditions of the simulated system with acceptable margins. An important aspect related with a nodalization is that it can be considered qualified when it has a geometric fidelity with the system, it reproduces the measured steady-state condition of the system, and it demonstrates satisfactory time evolution conditions (D'Auria et al., 1999). However, sometimes a nodalization qualified to simulate determined condition may not be suitable to simulate other type of situation being necessary modifications and re-qualification of the model.

RELAP5 steady-state calculation was performed at 50 kW. The temperature values at inlet and outlet of the TH channels 3, 8 and 13 calculated using RELAP5 can be verified in Table 4. The calculated values were compared with the available experimental data (inlet and outlet channel temperature) and with to the STHIRP-1 calculation data (outlet channel temperature) (Veloso, 2004).

Both codes presented predictions with good agreement respect the experimental data (Table 4). The error obtained by the RELAP5 calculation is a few overestimated (to channels 8 and 13) in com-

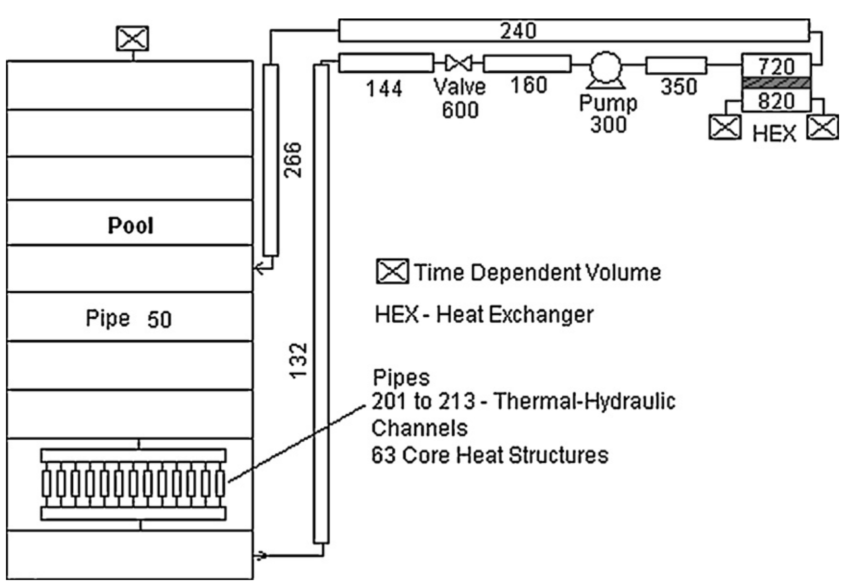


**Fig. 4.** IPR-R1 TRIGA nodalization using the RELAP5 model.

**Table 4**
Experimental and calculated results for 50 kW thermal power condition.

| TH channel | Outlet channel temperature (K) | | | | | Inlet temperature (K) | | |
|---|---|---|---|---|---|---|---|---|
| | Exp. | RELAP5 | Error (%)[a] | STHIRP-1 | Error (%)[a] | Exp. | RELAP5 | Error (%)[a] |
| 3 | 300.1 | 298.5 | 0.5 | 299.0 | 0.4 | 294.1 | 294.5 | 0.1 |
| 8 | 298.1 | 296.2 | 0.6 | 297.6 | 0.1 | 296.1 | 294.5 | 0.5 |
| 13 | 298.1 | 296.2 | 0.6 | 298.5 | 0.1 | 296.1 | 294.5 | 0.5 |

[a] Error = 100 × (calculation − experimental)/experimental.

parison with the maximum acceptable error suggested for coolant temperature (0.5%) by the RELAP5 users (D'Auria and Galassi, 1998). However, considering the error from the experimental data, the results found are perfectly acceptable for an initial validation of the model at 50 kW. Chromel–alumel calibrated thermocouples were used to collect the coolant temperature data and the measured values have a maximum error of ±1 °C (Veloso, 2004). STHIRP-1 code reached values of outlet temperature with minimum error demonstrating an excellent reproduction of the steady-state reactor behavior. It is convenient to remember that the STHIRP-1 code was developed specially according with the IPR-R1 TRIGA characteristics and the core region was modeled considering 104 TH channels against 13 TH channels in the present RELAP5 model. The 104 TH channels in the STHIRP-1 nodalization for the IPR-R1 core are shown in Fig. 6.

STHIRP-1 inlet temperature values are not available. The inlet temperature values calculated using the RELAP5 code are the same for the three considered channels. Refinement of the present RELAP5 model and sensitivity calculations will be performed in future works aiming to increase the number of TH channels and to investigate the actual effect of this in the calculations.

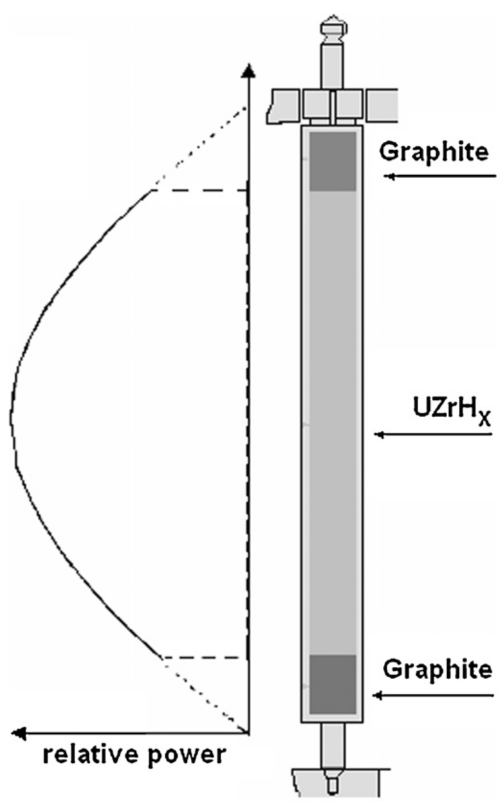


**Fig. 5.** Prediction of the axial power distribution function in a fuel element.

Fig. 7 shows the calculated time evolution for inlet and outlet temperature at 50 kW power, for the TH channel number 1. Such channel was chosen because it concentrates the HS with higher values of radial power. As it can be verified, after about 2000 s of calculation, the temperature reaches the steady-state condition having a value in good agreement with the experimental available data.

In addition, Fig. 8 provides the fuel and cladding temperature evolution for the heat structure 1 (HS-B1) in the channel 1 at the mid high. As it can be observed, these parameters are completely stable and the difference of temperature between them is approximately 30 K. The HS-B1 corresponds to the fuel element in the position B1 according to Fig. 2. Fig. 9 presents the time evolution for the HS-B1 fuel temperature at four different axial levels. As for the power, also the axial fuel temperature distribution follows the cosine profile function, reaching higher temperatures in the central parts of the element, as demonstrated in Fig. 10 for the case of the HS-B1.

### 3.2. *Transient calculation at 50 kW—forced recirculation off*

Several selected postulated initiating events (PIEs) for research reactors have been classified and summarized as follows (IAEA, 2005):

1. Loss of electrical power supplies;
2. insertion of excess reactivity;
3. loss of flow;
4. loss of coolant;
5. erroneous handling or failure of equipment or components;
6. special internal events;
7. external events;
8. human errors.

However, for TRIGA reactors, due the passive nature of the reactivity feedback during a temperature excursion, few PIEs would be applied since any increase in core temperature has a negative reactivity effect, causing a passive reduction in reactor power to limit a temperature excursion (IAEA, 2008).

The IPR-R1 is an inherently safe reactor. In spite of this, some perturbation situation may occur disturbing the normal reactor operation. For example, a condition of forced recirculation off caused by the recirculation pump failure was classified inside the event number 3 described before. Therefore, considering a fault of the forced recirculation system, an hypothetical transient has been investigated in this work. Measurements recently performed at 100 kW in the IPR-R1 reactor pool during the calorimetric power calibration technique has demonstrated that the average temperature-rise rate is about 4.8 °C/h (Mesquita et al., 2009). In such experiment, the reactor operated during a period of about 2.5 h with the forced cooling system switched off and with an indication of 100 kW at the linear neutronic channel (Mesquita et al., 2009).

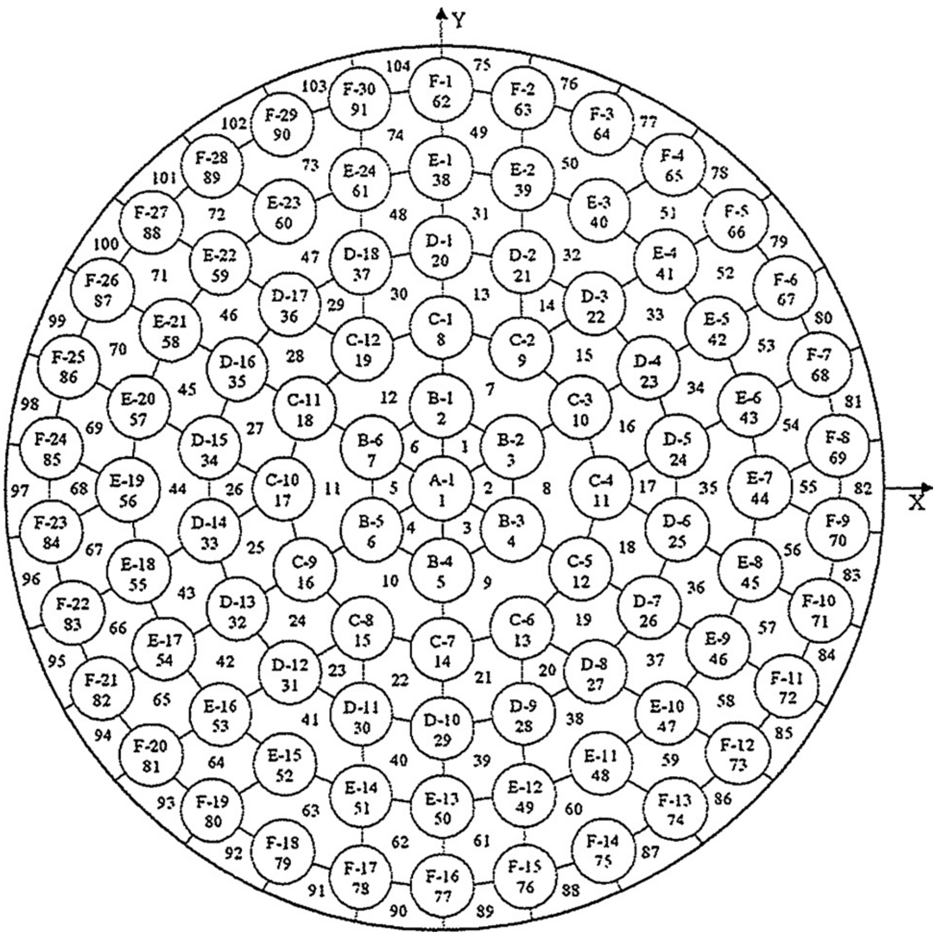


**Fig. 6.** TH sub-channels configuration of IPR-RI core using the STHIRP-1 code.

To perform the simulation, the valve in the primary system (number 600 in the RELAP5 nodalization) was closed at 3000 s of calculation after the system to reach a steady-state condition. After the beginning of the transient, the temperatures increase as a consequence of no energy removal from the core since the primary was off (see Fig. 11). The valve remained closed up to the end of the calculation.

The scram intervention was not considered in the simulation. In an actual situation, the reactor is automatically shutdown in cases of the primary or secondary flow rates are below the set point. After the beginning of the transient, fluid temperature increases gradually at a mean rate of about 30.0 °C/h, as it can be verified in Fig. 11. Void formation was not observed to appear at the TH channel investigated in spite of the temperatures rising.

The calculated rate was overestimated six times in comparison with the recent available experimental data (4.8 °C/h) at 100 kW. Model investigations will be performed to improve the calculation results and to re-qualify the nodalization to simulate adequately such transient. An idea is to include in the present nodalization the cross-flow model to the pool aiming to improve the heat removal

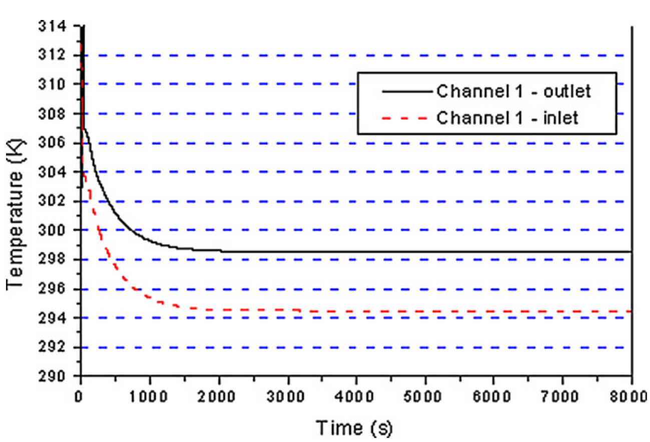


**Fig. 7.** Inlet and outlet coolant temperature for the channel 1 predicted using the RELAP5 model.

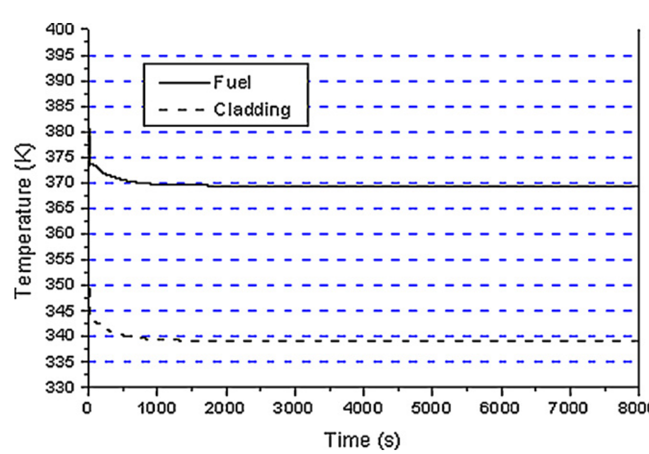


**Fig. 8.** HS-B1 cladding and fuel temperatures predicted using the RELAP5.

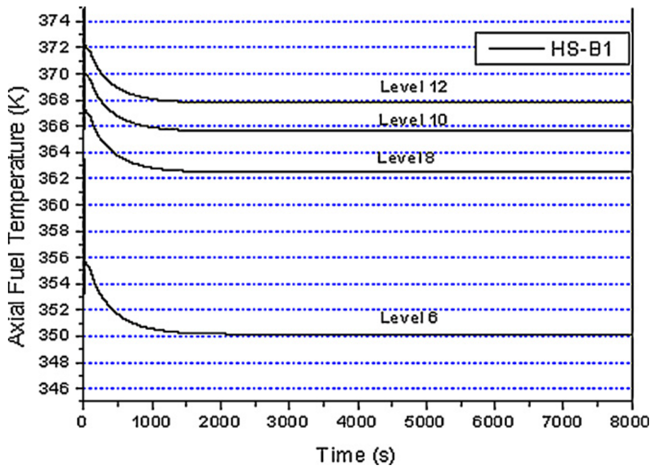


**Fig. 9.** Fuel temperature evolution at four levels to the HS-B1.

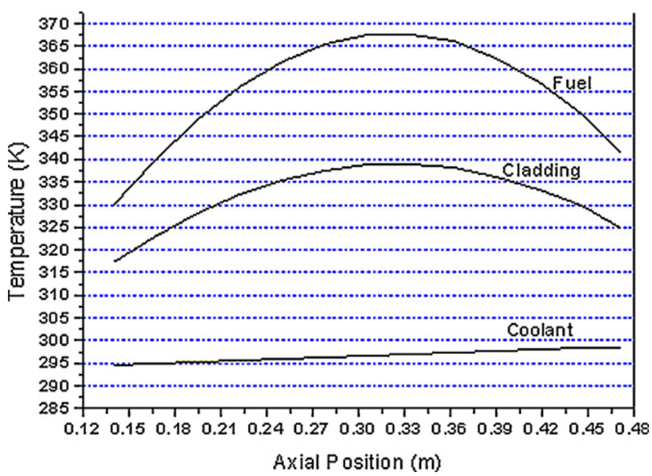


**Fig. 10.** Axial fuel, cladding and coolant temperature distribution.

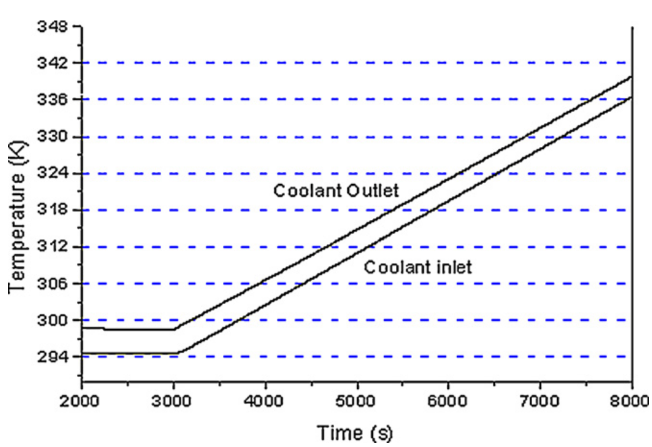


**Fig. 11.** Forced recirculation off at 3000 s.

from the core and also to perform simulations using the 3D neutron kinetic model present in the RELAP5-3D code.

## 4. Conclusions

In this work, a nodalization for the IPR-R1 TRIGA research reactor performed using the RELAP5/MOD3.3 code has been presented as a contribution to the assessment of such code for research reactor safety analysis. The nodalization was validated against experimental data from IPR-R1 steady-state conditions at 50 kW characterizing a code-to-data validation. The RELAP5 results have been also compared with some data obtained using the already validated STHIRP-1 thermal hydraulic code.

The results are in good agreement between the codes with little discrepancies which could be explained by the different empirical correlations embedded within each code. Effects on the physical parameters, such as velocity or energy must be accounted for through algebraic terms added to the conservation equations. These terms should be based on correlations deduced from experimental data for their representation or on models developed from sound physical principles, as it is the case for the STHIRP-1 code (Veloso, 2004). Some of the correlations used in RELAP5, however, are based on engineering judgment, due partly to the incompleteness of the science and partly to numerical stability requirements (US NRC, 2001).

The little discrepancies can be also related to the differences on the nodalization methodologies adopted for each code. To complete the process of validation for the present RELAP5 model, other parameters must be compared with experimental data. Moreover, calculations at several power levels are necessary to prove the nodalization effectiveness.

Future investigations will be performed to verify the effect of the number of TH channels in the calculation results. Furthermore, transient calculations must be performed as a second step in the code validation process. The forced recirculation off transient was investigated by the code simulations presenting results overestimated in comparison with the experimental available data. It was not considered, for this simple model, cross-flow or mass flow between the channels. In this way, we modeled the core considering only vertical cooling movement. For steady-state low power as for such system, this consideration is acceptable and the results presented were in good agreement with the experimental ones. However, it has been verified that it is necessary to consider the cross-flow model to predict in a more realistic way transient conditions.

If considering the three basic aspects necessary to qualify a nodalization for a system (geometric fidelity, reproduction of the measured steady-state conditions and satisfactory time evolution conditions), it is possible to conclude that the RELAP5 model presented in this work was qualified to the IPR-R1 TRIGA research reactor considering operation at 50 kW of power and steady-state conditions.

The results from this phase combined with the results from the future work will provide both experimental and numerical information, as well as detailed information about normal and off-normal transient phenomena that could occur in research reactors. Future investigations for the IPR-R1 TRIGA nodalization include a cross-flow model and also coupled neutron kinetic/thermal hydraulic analysis.

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