THERMAL HYDRAULIC SIMULATION OF THE TRIGA IPR-R1 RESEARCH REACTOR

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ABSTRACT

The safety analysis of research reactors includes simulations of selected cases classified by the International Atomic Energy Agency (IAEA), 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. A RELAP5 model verified for the IPR-R1 TRIGA research reactor was used here to study. A loss of coolant accident (LOCA) event was simulated. The obtained results demonstrate that it is necessary to consider also the neutronic feedback effects in the thermal hydraulic calculations to more realistically simulate this type of incident as it is shown in this work.

INTRODUCTION

The nuclear research reactors work at low power (maximum of about 100 MW) in relation to power reactors. Even so, specific features are necessary to ensure safe utilization of research reactors. Therefore, several codes have been used focusing special attention for research reactors safety analysis and valuation of specific perturbation plant licensing and operation of research reactors.

Loss of electrical power supplies, insertion of excess reactivity, loss of flow accident (LOFA), loss of coolant accident (LOCA), erroneous handling or failure of equipment or components, special internal events, external events and human errors are considered possible events in research reactors [1]. In this way, spite the IPR-R1 to be inherently safe, some situations can disturb its normal operation.

In this work, the thermal hydraulic RELAP5 code [2] was used to perform steady state and transient simulations of the IPR-R1 TRIGA research reactor. The RELAP5 system code was developed to simulate transient scenarios in power reactors, but it has been also applied to simulate research reactor operation conditions.

The RELAP5 model presented in this work has demonstrated to reproduce very well the steady-state conditions. Therefore, a hypothetical extreme case of LOCA was investigated using the code and the model developed. LOCA is a consequence of one of the following events: rupture of the primary coolant boundary, damaged pool, pump-down of the pool, failure of beam tubes or other penetrations. During the LOCA event, the internal natural circulation of the fluid is suppressed and an excessive core heat-up is expected to occur [3].

Particularly, in this work an extreme case of LOCA was investigated. The probability of this kind of accident in the IPR-R1 is reduced but must be considered. In the IPR-R1, the LOCA accident would occur due to a drastic crack in the pool bottom. For the LOCA simulation the automatic scram was not considered because the main aim was to verify the temperature evolutions after total loss of the coolant in the pool.

The IPR-R1 fuel is an Uranium and Zirconium Hydride alloy (U-ZrH_x). In TRIGA type reactors the fuel temperature limit is related to the dehydrogenation of the fuel and consequent tension in the cladding of the fuel.

The maximum temperature of the fuel to avoid the dehydrogenation is fixed by General Atomic as 1173.0 K for the U-$ZrH_{1.0}$ and 1423.0 K for the U-$ZrH_{1.6}$ [4]. Howerver, for the reactor operation, the fuel temperature limit is maintained at about 890.0 K that is the temperature for phase changing of U-$ZrH_{1.0}$ fuel [5].

NOMENCLATURE

ρ	[kg/m³]	Density
k	[Wm-¹K-¹]	Thermal conductivity
c_p	[kJ.kg-¹K-¹]	Specific heat
HS		Heat Structure
THC		Thermal Hydraulic Channel
V_m/V_f		(moderator volume/fuel volume) ratio
LOFA		Loss of Flow Accident
LOCA		Loss of Coolant Accident

Subscripts

U-ZrH_x	Mixture of zirconium hydride and Uranium
ZrH	Zirconium hydride

IPR-R1 – GENERAL DESCRIPTIONS

IPR-R1 is a research reactor type TRIGA (Training, Research, Isotope, General Atomic), Mark-I model, manufactured by the General Atomic Company and installed at Nuclear Technology Development Centre (CDTN) of Brazilian Nuclear Energy Commission (CNEN), in Belo Horizonte, Brazil. The reactor is housed in a 6.625 meters deep pool with 1.92 meters of internal diameter and filled with demineralized light water. A schematic reactor diagram is illustrated in the Figure 1.

The water in the pool has function of cooling, moderator and neutron reflector and it is also able to assure an adequate

radioactive shielding. The reactor cooling occurs predominantly by natural convection, with the circulation forces governed by the water density differences. To perform the heat removal generated in the core, the water of the pool is pumped 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 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. There are 59 fuel elements covered with aluminium and 4 fuel elements with stainless steel. The main thermal-hydraulic and kinetic characteristics of the IPR-R1 core are listed in [6].

The radial relative power distribution was calculated in preceding works using the WIMSD4C and CITATION codes 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.

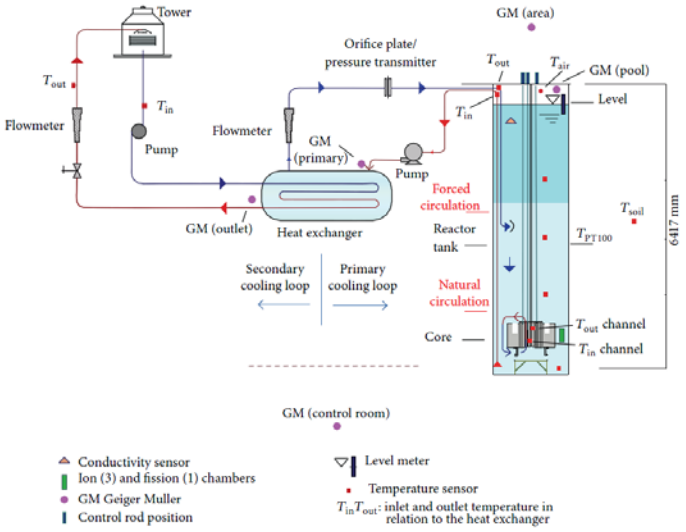

Figure 1 Schematic representation of the IPR-R1 [8]

Fuel Thermal Characteristics

The IPR-R1 core is composed by 59 fuel elements constituted by a cylindrical aluminium cladding and 4 fuel elements with cladding of stainless steel, both types filled with a homogeneous mixture of zirconium hydride and Uranium 20% enriched in ^{235}U isotope. The weight of hydrogen, uranium and zirconium differs in both types of elements. The mixture in the fuel (U-$ZrH_{1.0}$) with aluminium coating contains 8.0% uranium weight, 91% of zirconium and 1.0 % hydrogen. In the fuel element with stainless steel (U- $ZrH_{1.6}$) the percentages are 8.5 % in uranium, 89.9 % in zirconium and 1.6 % of hydrogen. The relation between the atoms of hydrogen to atoms of zirconium in the alloy is approximately 1.0 for fuel elements with aluminium cladding and 1.6 for elements with stainless steel cladding.

To adequately simulate this heat structure in the RELAP5 code, the fuel material thermal proprieties must be characterized.

1. U-ZrH_x density, $\rho_{U\text{-}ZrHx}$

The U-ZrH_x density is determined by:

$$\rho_{U-ZrH_x} = \frac{1}{w_U / \rho_U + w_{ZrH} / \rho_{ZrH}} \quad (1)$$

where w_U and w_{ZrH} are, respectively, the weight fractions of the Uranium and zirconium hydride in the fuel mixture; ρ_U is the Uranium density (19.07 g/cm^3) and ρ_{ZrH} is the zirconium hydride density. The zirconium hydride (ZrH_x) in relation to the hydrogen-zirconium percentage is given by the expressions purposed in [14]:

$$\rho_{ZrH} = \frac{1}{0.1541 + 0.0145x}; \quad x \le 1.6 \quad (2)$$

$$\rho_{ZrH} = \frac{1}{0.1706 + 0.0042x}; \quad x \ge 1.6 \quad (3)$$

To the mixture U-$ZrH_{1.0}$, substituting $w_U = 0.08$, $w_{ZrH} = 0.92$ e $x = 1.0$ in the preceding equations, it results in:

$$\rho_{U-ZrH_{1,0}} = 6.28\ \text{g/cm}^3.$$

This result agrees with the result given by the relation:

$$\rho_{U-ZrH_x} = 6.83 - 0.55x, \quad x \le 1.3 \quad (4)$$

for density of a ternary alloy with 8% in weight Uranium [15].
The density of the U-$ZrH_{1.6}$ mixture, with $w_U = 0.085$, $w_{ZrH} = 0.915$ and $x = 1.6$, is:

$$\rho_{U-ZrH_{1.6}} = 6.00\ \text{g/cm}^3,$$

where x is relation between the atoms of hydrogen to atoms of zirconium in the alloy.

2. U-ZrH_x Thermal Conductivity, *k*

The linear regression applied to the data presented by Ref. [16] to the thermal conductivity, *k*, of U-$ZrH_{1.0}$ with 8% in weight Uranium in relation to the temperature brings to the following approximation:

$$k_{U-ZrH_{1.0}}(T) = 22.872 - 4.3131x10^{-2}T + 1.1240x10^{-4}T^2 - 1.0039x10^{-7}T^3, \quad (5)$$

with temperature in ^{o}C and the conductivity in $Wm^{-1}K^{-1}$. The relation has validation in the temperature range of 72.0 ^{o}C ≤ T ≤ 410.0 ^{o}C.
The relation:

$$k_{U-ZrH_{1.6}}(T) = 17.58 + 0.0075T \quad (6)$$

is presented by [16] to the U-$ZrH_{1.6}$ thermal conductivity with 8.5% in weight Uranium. There is no indication of the temperature range validity to relation (6). However, *Simnad*

explains in Ref. [14] that the U-$ZrH_{1.6}$ thermal conductivity based in measures performed by *General Atomics Company* to thermal diffusivity and considering the density and specific heat, is independent of the Uranium percentage and the temperature; in this way, the value of 17.6 ± 0.8 $Wm^{-1}K^{-1}$ can be indicated to project calculations.

3. U-ZrH_x Specific Heat, c_p

The specific heat, c_p, of the U-ZrH_x, can be estimated by:

$$c_{p,U-ZrH_x} = w_U c_{p,U} + (1-w_U) c_{p,ZrH_x} \quad (7)$$

where w_U is the weight fraction of Uranium in the hydride. To the temperature range from 0 °C up to 668.0 °C, the Uranium specific heat can be approximated by:

$$c_{p,U}(T) = 0.1145 + 8.456\times10^{-5}T - 3.435\times10^{-8}T^2 + 1.692\times10^{-10}T^3 \quad (8)$$

with $c_{p,U}$ in $kJ.kg^{-1}K^{-1}$ and T in °C. The $ZrH_{1,0}$ specific heat is given by the following approximation, validated for the temperature interval of 50.0 to 525.0 °C:

$$c_{p,ZrH_{1,0}}(T) = 0.310 + 6.66\times10^{-4}T \quad (9)$$

Substituting the equations (8) and (9) in the equation (7) and being $w_U = 0.08$, results in the expression for the U-$ZrH_{1.0}$ fuel with 8% in weight Uranium:

$$c_{p,U-ZrH_{1,0}}(T) = 0.294 + 6.196\times10^{-4}T - 2.748\times10^{-9}T^2 + 1.354\times10^{-11}T^3 \quad (10)$$

The ZrH_x enthalpy, h, (in J/mol), can be expressed as [18]:

$$(h - h_{25^oC})_{ZrH_x} = a(x) + b(x)T + 0.03488T^2 \quad (11)$$

with T in °C, $a(x) = -882.95 + 370.18(1.65 - x)$ and $b(x) = 34.446 - 14.8071(1.65 - x)$.

Using the specific heat definition, $c_p = (\partial h/\partial T)_p$, we have:

$$c_{p,ZrH_x}(T) = b(x) + 0.06976T \quad (12)$$

In the case of U-$ZrH_{1.6}$, x = 1.6. Therefore, taking x = 1.6 and considering the molecular mass of the $ZrH_{1.6}$ as 92.8328, this equation becomes:

$$c_{p,ZrH_{1.6}}(T) = 0.3631 + 7.5146\times10^{-4}T, \quad (13)$$

with c_p in $kJ.kg^{-1}K^{-1}$. Substituting $w_U = 0.085$ and the equations (8) and (13) in the Eq. (7), we have the approximation for the specific heat to U-$ZrH_{1.6}$ with 8.5% in weight Uranium:

$$c_{p,U-ZrH_{1.6}}(T) = 0.3420 + 6.948\times10^{-4}T - 2.920\times10^{-9}T^2 + 1.438\times10^{-11}T^3 \quad (14)$$

This data were used to characterize the fuel material in the RELAP5 model for the IPR-R1 reactor. Also the data related with the gap and cladding materials were considered [17].

MODELLING USING THE RELAP5 CODE

The IPR-R1 nodalization is represented in the Figure 2. The reactor pool was modeled using two pipe components. The natural convection system and the primary loop circulation have been modeled.

To simulate the forced circulation, the pipe 132 was connected in the first volume of the pool using the single junction 131. The water returns to the pool coming into the volume 6 through the pipe 266. The pump 300 supplies the water circulation. In the nodalization the valve 325 is normally closed. It opens to simulate the LOCA transient forcing the cooling to follow to the TDV 327.

In the nodalization, each of the 63 fuel elements was modeled separately and was associated with the corresponding hydrodynamic pipes constituting 03 core thermal hydraulic channels (THC). Sensitivity tests were performed considering different number of channels as it will be described in the section "Model Verification".

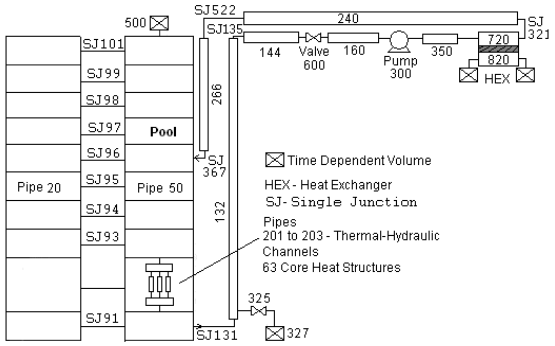

Figure 2 IPR-R1 nodalization in the RELAP5

Heat Structures

Modeling heat structures in the RELAP5 code is possible to predict heat conduction across the solid boundaries of the hydrodynamic volumes. Heat transfer can be modelled through fuel pins or plates, steam generator tubes, and pipe and vessel walls [2]. Thermal conductivities and volumetric heat capacities as functions of temperature can be input in tables. Finite differences are used to advance the heat conduction solutions.

The fuel pin of the IPR-R1 reactor was modelled as a heat structure in the RELAP5 code according with the Fig. 3. In the model, the fuel pin was divided axially in 21 parts, and radially it has 17 meshes divided as shown in the figure including fuel, gap and cladding regions. Two different types of cladding materials were considered being aluminium (59 elements) or stainless steel (4 elements).

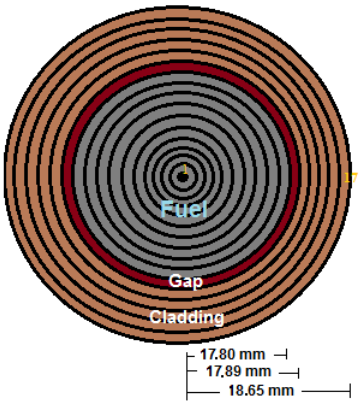

Figure 3 IPR-R1 fuel pin modelled in the RELAP5 (out of scale)

RELAP5 - RESULTS AND ANALYSES

Model Verification - Steady State and Transient Results

The RELAP5 steady state calculations were performed for models with 3, 13 and 91 core thermal hydraulic channels at 100 kW [9]. The models were verified using experimental available data [20]. In the Fig. 4 is presented a scheme of how the channels were defined for each type of core model. The points identified as E_a, E_b and E_c in the Fig. 4a represent, approximately, the positions were the experimental temperature data were taken [20]. In the Tab. 1, is presented a comparison among the temperature values in the outlet of the channels to the three core nodalizations in relation with the available experimental data. For the model with three channels, the comparison was performed with the average experimental data. Beyond the temperature, other parameters were compared with experimental data as mass flow and pressure with good agreement for all developed models. From Tab. 1, the error presented by the 91 THC model was lower in relation to the other models. However, the model with 3 THC was used to perform the calculations because it is simpler and then spends less computational time.

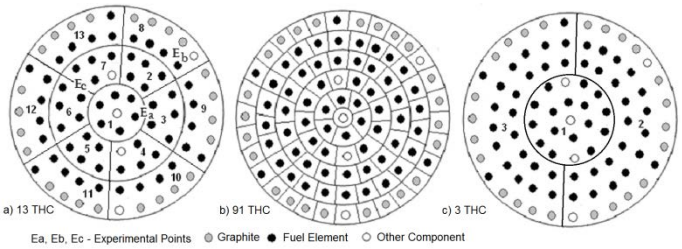

Figure 4 a) IPR-R1 modelled in the RELAP5 with 13 THC. Points E_a, E_b and E_c represent the experimental positions; b) model with 91 THC; c) model with 3 THC.

Steady State calculations were performed for the model with 3 THC. The steady state is reached at about 2500.0 s of calculation. There is an increase of the coolant temperature of approximately 4.9 K, as it was expected.

Table 1 Temperature in the outlet channels for four modelings in comparison with experimental data at 100 kW

Experimental position [20]	Coolant Temperature - Outlet of THC								
	Experimental value (K)	Error (K)	13 THC (K)	Error (%)*	91 THC (K)	Error (%)*	Exp. mean value (K)	3 THC (K)	Error (%)*
E_a	304.0	± 1.0	301.4	0.9	303.6	0.1	301.8	300.3	0.5
E_b	300.5		298.1	0.8	301.1	0.2			
E_c	301.5		300.9	0.2	301.8	0.1			

*Error = 100 x (calculation – experimental)/experimental

To verify the model with 3 THC at transient conditions, the event of stopping the forced coolant recirculation was simulated and the results were compared with experimental data. It may be caused by a pump failure, for example. In the experiment, the reactor operated during about 2.5 hours with the forced cooling system switched off and at 100 kW of power operation. The experimental measurements demonstrated an average temperature rise of about (4.8 ± 1.0) °C/h [21].

To perform the simulation using the RELAP5 the valve in the primary system (number 600 in the nodalization of the Fig. 2) was closed at 4000 s of calculation when the system was in steady state operation condition. The temperature increase in the core was verified (Fig. 5). The result obtained with the RELAP5 calculation was 4.70 ◦C/h, demonstrating very good agreement with the experimental available data.

Transient Results - LOCA

LOCA has been simulated using the RELAP5 model for the IPR-R1. After to reach steady state behavior at 100.0 kW of power operation, the transient started at 4000 s of calculation. To perform the simulation, the valve 325 (see Fig. 2) connected in the last volume of the pool was opened. The water in the pool began to decrease immediately. The coolant was forced to coming into the TDV 327 that is at atmospheric pressure. In the Figure 6 is shown the pool water level time evolution that is initially at about 6 meters from the bottom (black line). The water pressure in the pool bottom dropped reaching atmospheric level (Figure 6 – red line) as it was expected. Figure 7 shows the void fraction evolution at pool bottom. After about 150 s of calculation since the beginning of the LOCA, the core is practically all uncovered.

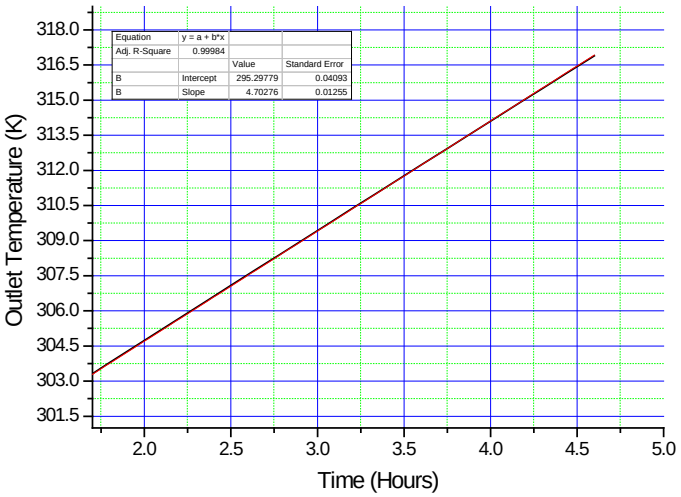

Figure 5 Core coolant temperature increasing after the closure of the forced recirculation in the IPR-R1 model

In the Fig. 8 is represented the fuel and cladding temperatures time evolution at the middle of the heat structure 205, that represents one of the fuel elements belonging to the THC 201 (number 1 in the Fig. 4). After the core is totally uncovered, the heat generated by the fuel is no more removed by convection being then realized only by radiation. As it can be verified in the Fig. 8, the fuel and cladding temperatures

increase drastically beyond the project limits (890 K). The calculation stops because the values reached are out of the temperature range for fuel and cladding materials. This drastic increase in the fuel temperature is a not expected behavior.

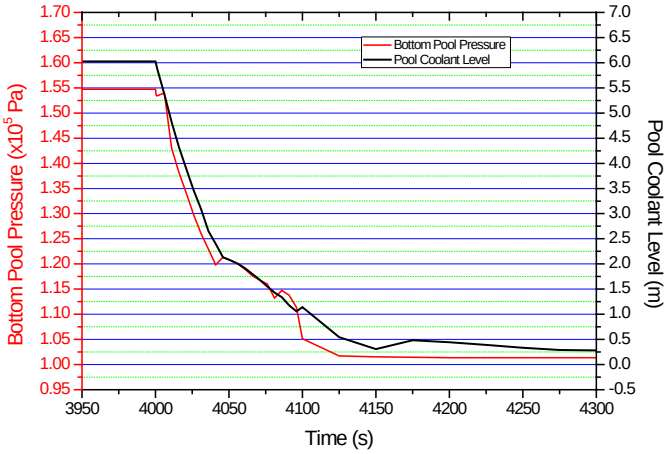

Figure 6 Water pool level (red line) and pressure at the pool bottom time evolution

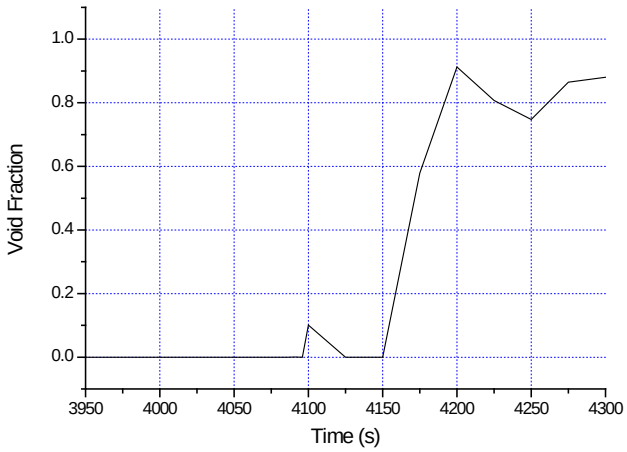

Figure 7 Pool bottom void fraction.

CONSIDERATION OF THE NEUTRON KINECTICS EFFECTS

During the LOCA event, an excessive core heat-up occurs. This fast change in the core temperature will cause changes in the fuel macroscopic cross sections, strongly influencing the reactivity. Such process, known as *feedback effects*, will affect the core criticality having an important role in the reactor operation [19]. As verified firstly, the point kinetics calculation adopted by the RELAP5 was not capable to predict the reactor behaviour after the considered event because in such type of calculation the cross sections feedback effects are not considered.

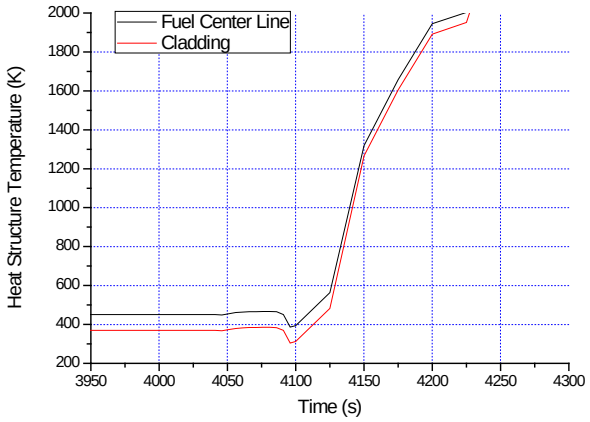

Figure 8 Cladding and fuel time evolution temperatures in the middle of a fuel element in the region of the THC 201

Therefore, to try to simulate adequately this transient, the neutron kinetics effects in a 3D calculation coupled with thermal hydraulic effects have been considered. In this way, the neutron kinetics PARCS code [10] was coupled to the RELAP5 and the transient simulation was performed again. In the RELAP5-PARCS coupling calculation, PARCS [10] makes use of the moderator temperature and density and of the fuel temperature calculated by RELAP5 to evaluate the appropriate feedback effects in the macroscopic neutron cross sections and calculate the power for each neutronic node in the core. Likewise, RELAP5 takes the space-dependent power calculated in PARCS and solves the heat conduction in the core heat structures for the corresponding nodes.

Coupling between Thermal Hydraulic and Neutronic Codes

The neutron kinetic code PARCS was coupled to the RELAP5 code. The coupling process between RELAP5 and PARCS codes is done through a parallel virtual machine (PVM) environment, using an adequate association among thermal hydraulic and neutronic nodes. To perform the calculation it is necessary to provide neutron macroscopic cross sections to the PARCS code that are calculated using a cell code as WIMSD-5B program.

To simulate the IPR-R1 core in the PARCS code, a Cartesian geometry was considered [13]. The modelling to Cartesian geometry is shown in Fig. 9 (left side), where the number 2 represents fuel region and the number 1 represents the reflector region. As an example, Fig. 9 (right side) shows the fuel compositions (2, 3, 4, 5, 6, 9), rod regions (7), the central thimble (10) and reflector regions (1) for the axial planar region number 6. As there are 169 nodes for each plane and there are 21 axial planes, then the core has been modelled with 3549 neutron kinetic nodes. Each PARCS neutronic node is connected with a thermal hydraulic volume in the RELAP5 input characterizing a 3D reconstruction of the core.

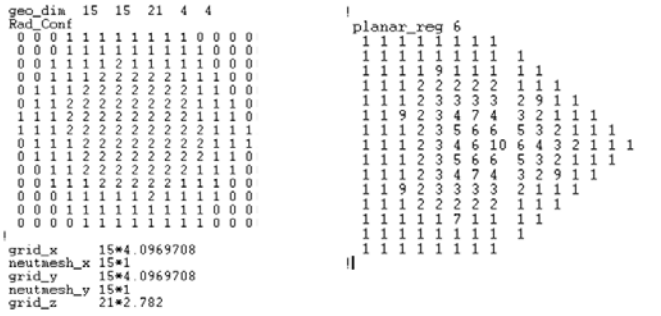

Figure 9 Left side: the IPR-R1 core representation in the PARCS code using Cartesian geometry; Right side: the axial planar region number 6 in the PARCS model

Macroscopic Cross Section Generation

The cross section libraries were generated by WIMSD-5B code (Winfrith Improved Multigroup Scheme) [11, 12] which is a general lattice cell program that uses transport theory to calculate flux as a function of energy and position in the cell.

In this work, geometry, position, composition and keeping the V_m/V_f (moderator volume/fuel volume) ratio were considered to define the cells. As output, WIMSD-5B code provides the diffusion coefficient and the macroscopic cross sections that are necessary for the neutronic code to perform the power calculation considering the feedback effects. The cross section sets generated by WIMSD-5B code were included in the PARCS model. Data as the scattering, absorption, fission, cross sections, and diffusion coefficient were inserted in PARCS input.

The parameters used to calculate the macroscopic cross sections, such as coolant and fuel temperatures, must represent situations of steady state and accidents. To IPR-R1 reactor, coolant temperature ranged from room temperature to the water saturation temperature at the pressure of 1.5 bar, which corresponds to 384.54 K. The fuel temperature was determined using the same criteria as in setting the coolant temperature. Room temperature was used as the lower limit, and 890.0 K was considered as the upper limit. This last value corresponds to the temperature for phase changing of U-$ZrH_{1.0}$. This represents a limiting safety factor of reactor.

RELAP5-PARCS COUPLING TRANSIENT RESULTS

The RELAP5-PARCS coupling was verified to the IPR-R1 steady state calculation in a previous work [13]. The LOCA event was performed again considering coupled calculation and the power operation of 100 kW as in the first case where it was considered only the thermal hydraulic calculation. The time of the LOCA occurrence was the same and after about 150 s of the transient beginning, the core was uncovered. In the Figure 10 is presented the behavior of the cladding and fuel temperatures for the coupled calculation. It is possible verify that in the coupled calculation also the temperatures reached the safety limits but there is an initial tendency to stabilization due the feedback effects in the cross sections. However, as occurred in the first case, the calculation stopped few time after the beginning of the event because of the extreme thermal conditions.

In the Figure 11 is possible to compare the RELAP5 stand-alone calculation with RELAP5-PARCS coupled calculation for fuel temperatures in the middle of a fuel element. As it is shown, the temperature increase in the RELAP5 point kinetics calculation is much higher than in the case of RELAP5-PARCS coupled calculation using neutron diffusion equation in the kinetics calculation. This fact actually confirms the need of to use feedback in the cross sections during an extreme event as the LOCA.

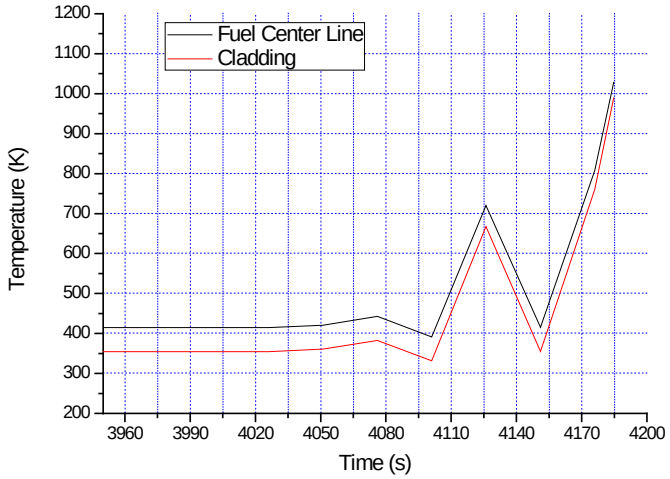

Figure 10 Cladding and fuel temperatures in the middle of a fuel element in the region of the THC 201– coupled calculation

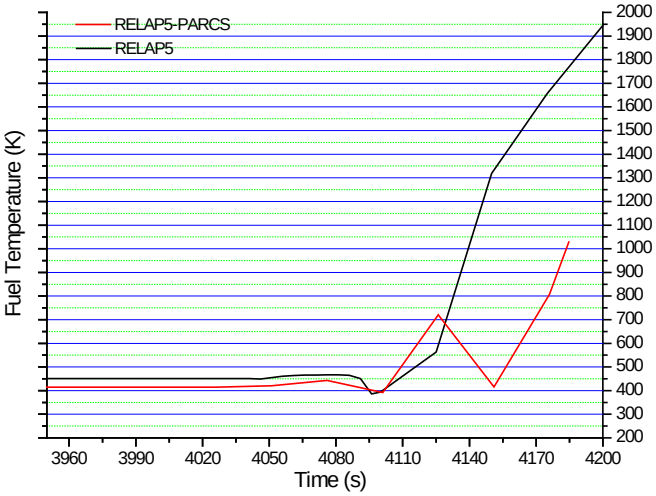

Figure 11 Fuel temperature in the middle of a fuel element in the region of the THC 201– comparison between RELAP5 stand-alone calculation and RELAP5-PARCS coupled calculation

CONCLUSION

In this work, an extreme case of LOCA was simulated for a research reactor. The calculations were firstly performed using only the thermal hydraulic code RELAP5. The model was previously verified in comparison with experimental data from the plant. However the high increase in the fuel temperature

demonstrated that for this type of event is essential to consider the neutronic feedback effects in the cross sections. Then, the transient was simulated again considering a thermal hydraulic/neutron kinetics coupled calculation. The coupled results demonstrate that when considering the neutronic effects in the power calculations, the fuel temperature increase less drastically than in the case of only thermal hydraulic calculation. More investigations are needed to better adjust the model. One problem that is necessary to study is the lack of "upscattering" calculation by the WIMS code during the process of the macroscopic cross sections generation. This is an important factor in the evaluation of transient situations in the case of TRIGA type reactors that have hydrogenated fuel.

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REFERENCES

[1] IAEA, Safety of Research Reactors. IAEA Safety Standards Series, Nº NS-R-4, IAEA, 2005, Vienna.

[2] US NRC, 2001. RELAP5/Mod3.3 Code Manuals, Idaho National Engineering Laboratory.

[3] Hamidouche, T. and Si-Ahmed, El-K., Analysis of loss of coolant accident in MTR pool type research reactor, *Progress in Nuclear Energy*, Vol. 53, 2011, pp. 285-289.

[4] Gulf General Atomic, 1970, Safeguards summary report for the New York University TRIGA Mark I reactor, Gulf General Atomic, San Diego, CA, GA-9864.

[5] Veloso, M. A., Avaliação termo-hidráulica do reator TRIGA IPR-R1 a 250 kW. CDTN/CNEN, 2005, Nota Interna NI-EC3-05/05.

[6] Reis, P. A. L., Costa, A. L., Pereira, C., Veloso, M. A. F., Mesquita, A. Z., Soares, H. V., Barros, G. P., Assessment of a RELAP5 model for the IPR-R1 TRIGA research reactor, *Annals of Nuclear Energy*, vol. 37, 2010, pp. 1341-1350.

[7] Dalle, H. M., Pereira, C., Souza, R. M. G. P., Neutronic calculation to the TRIGA IPR-R1 reactor using the WIMSD4 and CITATION codes, *Annals of Nuclear Energy*, vol. 29, 2002, pp. 901-912.

[8] Mesquita, A. Z., Costa, A. L., Pereira, C., Veloso, M. A. F. and Reis, P. A. L., Experimental investigation of the onset of subcooled nucleate boiling in an open-pool nuclear research reactor, *Journal of ASTM International*, vol. 8, 2011, pp. 1-9.

[9] Reis, P. A. L., Costa, A. L., Pereira, C., Silva, C. A. M., Veloso, M. A. F., Mesquita, A. Z., Sensitivity analysis to a REAP5 nodalization developed for a typical TRIGA research reactor, *Nuclear Engineering and Design*, Vol. 242, 2012, pp. 300-306.

[10] Downar, T. et al., PARCS v 2.7 U.S. NRC Core Neutronics Simulator, User Manual, School of Nuclear Engineering Purdue University, 2006.

[11] WIMSD-5B, 2007 - RSICC Computer Code Collection WIMSD-5B.12 – "Deterministic Code System for Reactor-Lattice Calculations" – *Oak Ridge National Laboratory*, 2007.

[12] Deen, J. R. and Woodruff, W. L., WIMS-ANL User Manual, *Argonne National Laboratory*, Argonne, Illinois, 2004.

[13] Reis, P. A. L., Costa, A. L., Pereira, C., Veloso, M. A. F., Scari, M. E., Miró, R., Verdú, G., Transient Cases Analyses of the TRIGA IPR-R1 using Thermal Hydraulic and Neutrons Kinetic Coupled Codes, *Proceeding of the International Nuclear Atlantic Conference – INAC*, São Paulo, Brazil, October 4-9, 2015.

[14] Simnad, M. T., The U-ZrHx alloys: its properties and use in TRIGA fuel, *Nuclear Engineering and Design*, 64, 1981, pp. 403-422.

[15] General Atomics Company, GA-9864, Safeguards Summary Report for the New York University TRIGA Mark I Reactor, San Diego, 1970.

[16] Wallace, W. P. and Simnad, M. T., Metallurgy of Triga fuel elements. General Atomic, San Diego, CA, 1961 GA-1949.

[17] Reis, P. A. L., Estudos Neutrônicos e Termo-Hidráulicos para o Desenvolvimento de uma Metodologia de Acoplamento entre Códigos Aplicada ao Reator de Pesquisa TRIGA IPR-R1, Doctor Thesis, Nuclear Engineering Department, Universidade Federal de Minas Gerais, 2013.

[18] Simnad, M. T., Foushee, F. C., West, G. B., Fuel elements for pulsed TRIGA research reactors, *Nuclear Technology*, 28, 1976, pp. 31-56.

[19] Duderstadt, J. J. and Hamilton, L. J., Nuclear Reactor Analysis, Ed. John Wiley & Sons, Inc., USA, 1976.

[20] Veloso, M. A. F., 2004. Análise Termofluidodinâmica de Reatores Nucleares de Pesquisa Refrigerados a Água em Regime de Convecção Natural", Doctor Thesis, Universidade Estadual de Campinas, São Paulo, Brasil, 2004.

[21] Mesquita, A. Z., Rezende, H. C., Souza, R. M. G. P., 2009. Thermal power calibrations of the IPR-R1 TRIGA nuclear reactor. In: Proceedings of the 20th International Congress of Mechanical Engineering, COBEM 2009, November 15–20, Gramado, Brazil.