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THERMAL HYDRAULIC SIMULATION OF THE TRIGA IPR-R1 USING THE RELAP5-3D

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Abstract. *Several transient events are possible to occur in research reactors: loss of electrical power supplies, insertion of excess reactivity, loss of coolant, erroneous handling or failure of equipment or components, special internal events, external events and human errors. In spite of the IPR-R1 to be inherently safe, situations that can disturb the normal reactor operation are possible to occur. The RELAP5-3D model presented in this work demonstrated to reproduce very well the IPR-R1 steady state and transient conditions. In addition to the verification process of the modeling, two transient events were investigated and the calculation results have been compared with available experimental data. One of the investigated events was the forced recirculation off that may be caused by the recirculation pump failure bringing the reactor to operate without forced recirculation conditions. In the other simulated case the pump velocity is brought to zero in a short time period and the reactor remains only with a minimum forced recirculation. The measurements have demonstrated an average temperature rise rate of about 4.8 °C/h after the transient for the first case. The simulation results demonstrated good agreement with the experimental available data as it is shown in this work.*

Keywords: *RELAP5-3D, TRIGA IPR-R1, Natural Circulation*

1. INTRODUCTION

Safety evaluations of research reactors are mainly performed by deterministic methods. The best estimate method provides a realistic simulation of a physical process to a level commensurate with the currently known data and knowledge of the phenomena concerned and it gives a good view of the existing margins or limits on research reactor transient scenarios in relation to the safety analysis (IAEA, 2008). The use of a best estimate code is essential for best estimate analysis including simulations of complex scenarios in nuclear power plants. Such simulations have been improved by the utilization of best estimate coupled thermal-hydraulic (TH) and neutron kinetics (NK) system codes thanks also to development of the computer technology and new calculations methodology making possible to perform transport calculation schemes with accurate solutions. The TH-NK coupling technique consists in incorporating three-dimensional (3D) neutron modeling of the reactor core into system codes mainly to simulate transients that involve asymmetric core spatial power distributions and strong feedback effects between neutronic and reactor thermal-hydraulic behaviors (Costa et al., 2008).

The TH-NK coupling technique was initially developed and used to simulate the behavior of power reactors. However, several coupling methodologies are now being applied for research reactors, mainly for TRIGA type, as can be verified for example in (Reis et al., 2012; Reis et al., 2010; Kriangchaiporn et al., 2010; Marcum et al., 2010; Feltus and Miller, 2000). 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 ñinherent safetyò. In spite of this, some situations may occur disturbing the normal reactor operation. In the present work, the IPR-R1 TRIGA reactor installed in Brazil (in operation since 1960) has been modeled for RELAP5/MOD3.3 and RELAP5-3D codes with the aim of to reproduce the measured steady state as well as transient conditions.

The validation of a code modeling for determined system implicates that the model reproduces the measured steady state conditions of the system with acceptable margins. The nodalization may 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

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satisfactory time evolution conditions (DôAuria 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. Sensitivity analysis including systematic variations in code input variables or modeling parameters, must be used to help identify the important parameters necessary for an accident analysis by ranking the influence of accident phenomena or to bound the overall results of the analysis. Results of experiments can also be used to identify important parameters (Reis et al., 2012).

The main transient events possible to occur in research reactors are loss of electrical power supplies, insertion of excess reactivity, loss of coolant, erroneous handling or failure of equipment or components, special internal events, external events and human errors (IAEA, 2008; Adorni et al., 2005; Hawley and Kathren, 1982).

In a general way, the range of conditions for which a research reactor is designed constitutes its design basis where a number of unintended events are considered. According to the probability of its occurrence and potential consequences, such an event may be classified as an anticipated operational occurrence or a design basis accident (DBA). An accident occurring outside the design basis is called a beyond design basis accident (BDBA) as, for example, the degradation of the reactor core (IAEA, 2008). In this way, safety analyses are used to demonstrate the safe operation of the reactor and how the design of the facility and the related operational procedures will contribute to the prevention and mitigation of accidents. It includes analyses of the response of the reactor to a range of PIEs (postulated initiating events), that refers to an unintended event including an operating error, equipment failure or external influence.

1.1 Main characteristics of the TRIGA IPR-R1

The IPR-R1 is a reactor type TRIGA of Mark-I model. It is installed at Nuclear Technology Development Centre (CDTN) of Brazilian Nuclear Energy Commission (CNEN), in Belo Horizonte, Brazil. It presents low power, low pressure, for application in research, training and radioisotopes production. The reactor is housed in a 6.625 meters deep pool with 1.92 meters of internal diameter and filled with light water which has function of cooling, moderator, neutron reflector and 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. All the 63 fuel elements are 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 aluminum and 4 fuel elements with stainless steel. The main thermal-hydraulic and kinetic characteristics of the IPR-R1 core are listed in (Reis et al., 2010). The radial relative power distribution (Fig. 1) was calculated in preceding works using the WIMSD4C and CITATION codes and also experimental data (Dalle et al., 2002). 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. IPR-R1 works at 100 kW but it will be briefly licensed to operate at 250 kW. Fig. 1 shows also the six core concentric rings (A, B, C, D, E, F).

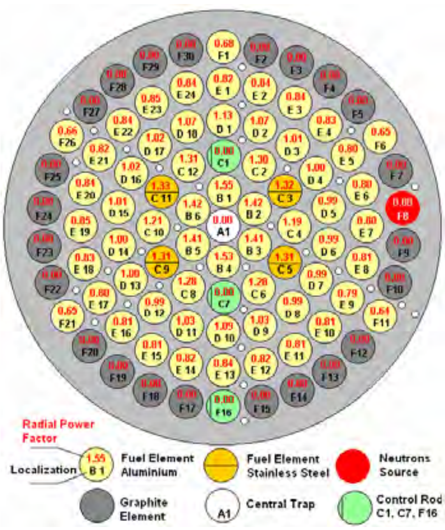

Figure 1. Core radial relative power distribution.

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2. THE RELAP5-3D CODE

The most prominent attribute that distinguishes the RELAP5-3D code from the previous versions is the fully integrated, multi-dimensional TH (Thermal-Hydraulic) and NK (Neutron-Kinetic) modeling capability (RELAP5-3D, 2005). There are two options for the computation of the reactor power in the RELAP5-3D code. The first option is the point reactor kinetics model that was implemented in previous versions of RELAP5. The second option is a multi-dimensional neutron kinetics model based on the NESTLE code developed at North Carolina State University (RELAP5-3D, 2005). RELAP5-3D was modified to call the appropriate NESTLE subroutines depending upon the options chosen by the user. The neutron kinetics model in NESTLE and RELAP5-3D uses the few-group neutron diffusion equations. Two or four energy groups can be utilized, with all groups being thermal groups if desired. Core geometries modeling include Cartesian and hexagonal. Core symmetry options are available, including quarter, half and full core for Cartesian geometry and one-sixth, one-third and full core for hexagonal geometry.

3. MODELING

The reactor pool was modeled using two pipe components, each one composed by ten volumes. The volumes of each pipe of the pool are equivalently connected between them by single junctions (SJ) characterizing a pool cross-flow model, as it can be verified in the nodalization represented in a general way in the Fig. 2. A time dependent volume (TDV) was used to simulate the atmospheric pressure on the pool surface (identifier 500). 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.

To simulate the forced circulation, the pipe 132 was connected in the first volume of the pool using a single junction. The water returns to the pool coming into the volume 6 through the pipe 266. The pump 300 supplies the water circulation.

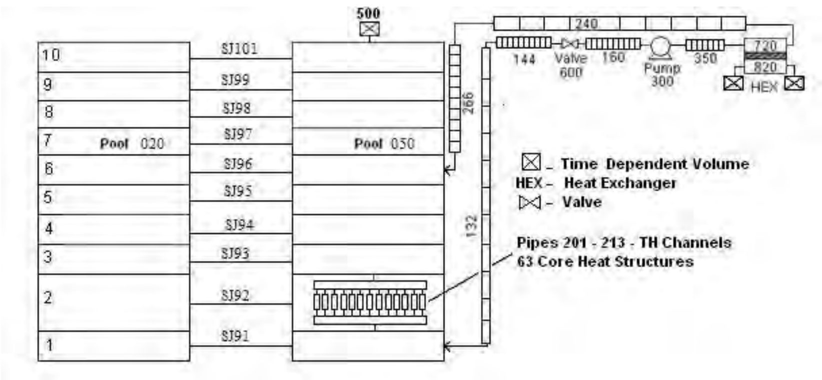

Figure 2. IPR-R1 ï Modeling using the RELAP5.

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 (201ï213), as can be seen in the planar core representation at Fig. 3.

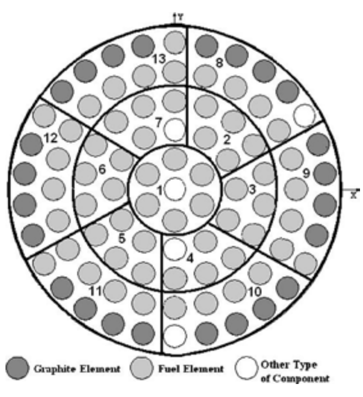

Figure 3. Core nodalization in RELAP5 showing the 13 TH channels.

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4. STEADY STATE RESULTS

The RELAP5/MOD3.3 and the RELAP5-3D steady state calculations have been performed for the IPR-R1 operating at 100 kW. The point kinetics model was used in both simulations. The axial power distribution was calculated considering a cosine profile. The RELAP5/MOD3.3 model for the IPR-R1 has been validated in a previous work against experimental data and also STHIRP-1 code results (Reis et al., 2010). It was verified that both models present the same time behavior for parameters as temperature, pressure and mass flow rate. As an example, Fig. 4 shows the coolant temperature at inlet and outlet of the channel 1 for both codes, with time evolutions very approximated each other. As it can be observed, after about 3000 s of calculation, both codes reach the steady state condition. As the RELAP5/MOD.3.3 is a validated model (Reis et al., 2012), it is possible to consider the RELAP5-3D model verified for IPR-R1 steady state operation at 100 kW.

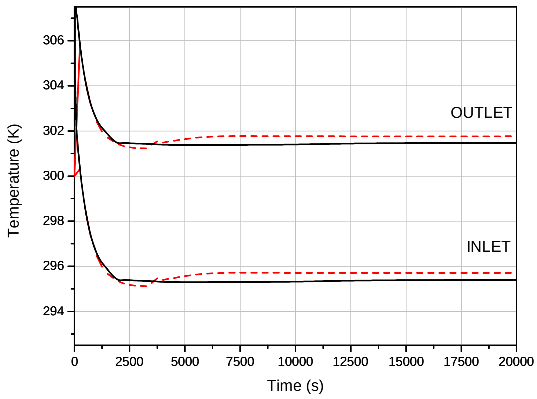

Figure 4. Coolant temperature evolution for channel 1 of the IPR-R1 at 100 kW of power operation for RELAP5-3D (dash red line) and RELAP5/MOD3.3 (straight black line) models.

5. TRANSIENT RESULTS

5.1 Forced recirculation off

In spite of the IPR-R1 to be inherently safe, situations that can disturb the normal reactor operation are possible to occur. Several experimental studies have been performed in the IPR-R1 reactor to ýnd out the core thermal power, the temperature distribution as a function of the reactor power under steady-state conditions, the þow distribution in the core coolant channels, the heat transfer coefýcient on the heated surface and a prediction of critical heat þux (Mesquita et al., 2007).

In one of such experiments, the reactor operated during about 2.5 hours with the forced cooling system switched off and with an indication of 100 kW at the linear neutronic channel. The measurements have demonstrated an average temperature-rise rate of (4.8 ± 0.2) °C/h (Mesquita et al., 2009). This transient event was investigated using the codes and the results have been compared with available experimental data of the forced recirculation off. The transient may be caused by the recirculation pump failure, bringing the reactor to operate in natural circulation conditions.

To perform the simulation using the RELAP5/MOD3.3, the valve 600 (in the nodalization) of the primary system was closed at 3000 s after the system to reach steady state condition. After the beginning of the transient, the core temperature increases as consequence of no heat removal from the pool since the primary was off as it is shown in the Fig. 5. The coolant temperature increases with a rate of 4.74 °C/h demonstrating very good agreement with the experimental available data and, therefore, this model was verified for this kind of transient case (Reis et al., 2012).

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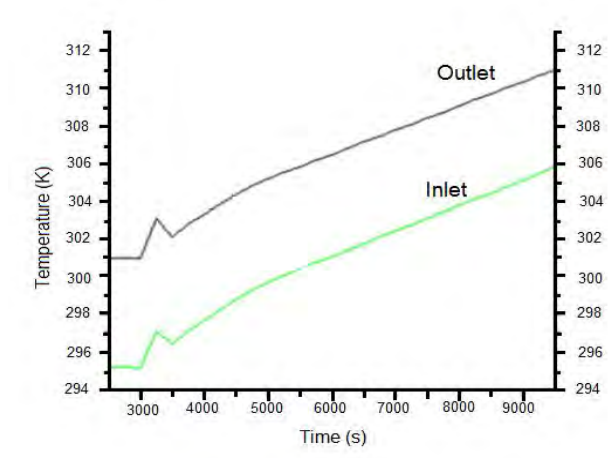

Figure 5. Inlet and Outlet core coolant temperature predicted by RELAP5/MOD3.3 after forced recirculation off at 3000 s.

Transient data from RELAP5-3D could not be obtained due problems during the calculation after the closing of the valve 600. Efforts have been performed to investigate the nature of such problems that seem to be connected with time step processing.

5.2 Pump stopping transient

It was simulated a similar case of that described in the section 5.1 using the RELAP5-3D to perform the pump stopping event. In this case, the pump velocity is brought to zero in a short time period and the reactor remains with natural circulation. Such case was performed also with the RELAP5/MOD3.3 model to compare the results that can be verified in the Fig. 6. The illustration presents the core inlet and outlet temperature increasing after the decrease of the pump velocity beginning at 4000 s of calculation to RELAP5-3D and RELAP5/MOD3.3 models. As it can be verified both models present the same behavior. RELAP5/MOD3.3 presents a temperature-rise rate of 2.82 °C/h while the RELAP5-3D presents a rate of 3.00 °C/h. The results agree very well between them. This value of about 3.00 °C/h is less than that found in the first case (section 5.1), and it was expected since there is a residual mass flow in the recirculation line (in both calculations) when the decrease of pump velocity is considered. Therefore, in this case, the residual mass flow that continues to circulate is responsible for a higher heat removing from the core and, consequently, the temperature-rise rate decreased. Differently, in the valve trip simulation (section 5.1), the mass flow effectively falls to zero in the recirculation coolant line.

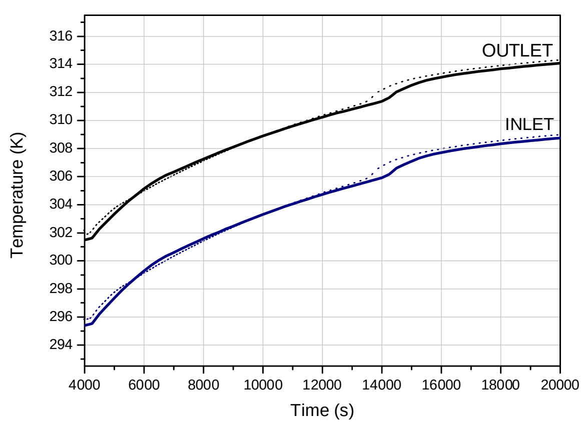

Figure 6. Core inlet and outlet temperatures increasing after the decrease of the pump velocity beginning at 4000 s of calculation at 100 kW for RELAP5-3D (dash line) and RELAP5/MOD3.3 (straight line) models.

6. CONCLUSIONS

In this work, a nodalization for the IPR-R1 TRIGA research reactor for RELAP5/MOD3.3 and RELAP5-3D codes has been presented as a contribution to the assessment of these codes for research reactor safety analysis. The verification of the RELAP5-3D model was performed comparing their calculation results with the RELAP5/MOD3.3 model at 100 kW of power operation.

The steady state results of both codes showed good agreement between them. Good result has been obtained for transient simulation using RELAP5/MOD3.3 in comparison with experimental data as it was described in the section 5.1. However, problems during the transient calculation using RELAP5-3D demonstrated the necessity of a careful investigation to find a more adequate model for RELAP5-3D.

In the other hand, the transient that simulates the pump stopping was performed and the results agree very well between both codes. Therefore, it is possible to consider the verification of the RELAP5-3D model for IPR-R1 for steady state and transient cases. The future activity will consider the application of the multi-dimensional neutron kinetics model present in RELAP5-3D to verify the reactivity feedback effects for transient events.

7. ACKNOWLEDGMENTS

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