INSTALATION, VALIDATION AND UTILIZATION OF THE RELAP5 CODE FOR THERMAL HYDRAULIC ANALYSES IN THE NUCLEAR ENGINEERING DEPARTMENT OF THE UFMG

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Abstract. *The RELAP5 is a system code developed at the Idaho National Environmental and Engineering Laboratory for thermal hydraulic analysis of nuclear reactors. RELAP5 code is widely used for safety analysis studies of commercial nuclear power plants. However, recent studies with the RELAP5 code have shown significant capabilities for analysis of nuclear reactor research systems. As a contribution to the assessment of RELAP5/3.3 for research reactor safety analysis, this work presents steady state calculation results performed by RELAP5 for the TRIGA IPR-R1 research reactor model. The code has been installed in the Nuclear Engineering Department (DEN) of the UFMG and has been validated against the TRIGA IPR-R1 research reactor located at the Nuclear Technology Development Center (CDTN) in Brazil. The TRIGA IPR-R1 is a 250 kW, light water moderated and cooled, graphite-reflected, open-pool type research reactor.*
Steady-state thermal hydraulic analysis of TRIGA IPR-R1 research reactor has been carried out. This paper presents the development and the validation of the TRIGA IPR-R1 model for the RELAP5 code using experimental data and also calculation data from the STHIRP-1(Research Reactors Thermal Hidraulic Simulation) cod developed at the DEN. As a result of this initial study, the TRIGA IPR-R1 nodalisation showed to be representative of the experimental data reactor behavior.

Keywords: *RELAP5, STHIRP-1, TRIGA IPR-R1*

1. INTRODUCTION

Currently, the enlarged commercial exploitation of nuclear research reactors has increased the consideration to their corresponding safety issues as can be verified, for example, in the works of Adorni (2007), Bokhari *et al.* (2002), Khedr *et al.* (2005) and Khater *et al.* (2007).

In general, the purpose of nuclear research reactors is not for energy generation; the maximum power generated within didn't exceed 100 MW. They are commonly devoted for generation of neutrons for different scientific and social purposes. However, high power densities are involved in the core and specific features are necessary to ensure safe utilization of these installations (Adorni, 2007). In addition to their particular characteristics, including large variety of designs, wide range of powers, different modes of operation and purposes of utilization, special attention should be focused for their safety aspects. In this way, several system codes, as the RELAP5 code, have been used for system safety analysis and valuation of specific perturbation plant processes.

The thermal-hydraulic RELAP5 system code was developed to simulate transient scenarios in power reactors such as PWR and BWR. However, some recent works (Khedr *et al.*, 2005, Antariksawan *et al.*, 2005, Adorni, 2007, etc) have been performed to access the applicability of the code to research reactors operating conditions (low pressure, mass flow rates, power, etc).

In this paper, the development and the validation of the TRIGA IPR-R1 model for the RELAP5 code have been presented. The validation has been performed using experimental data and also calculation data from the STHIRP-1 code (Veloso, 2004), developed at the DEN. As a result of this initial study, the TRIGA IPR-R1 nodalisation has shown to be representative of the experimental data reactor behavior. Steady-state thermal hydraulic analysis of TRIGA IPR-R1 research reactor has been carried out.

1.1. The RELAP5 Code

The 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. Specific applications have included simulations of transients in LWR systems such as loss of coolant, anticipated transients without scram (ATWS), and operational transients such as loss of feedwater, loss of offsite power, station blackout, and turbine trip.

However, some recent works have been performed to access the applicability of the code to research reactors operating conditions (low pressure, mass flow rates, power, etc).

The development of the code began about in 1970 and continues at present time. The code is based on a non-homogeneous and non-equilibrium model for the two-phases that is solved by a fast, partially implicit numerical scheme to allow economical calculation of system transients.

The code includes many generic component models from which general systems can be simulated. The component models include pumps, valves, pipes, heat releasing or absorbing structures, reactor point kinetics, electric heaters, jet pumps, turbines, separators, accumulators, and control system components. In addition, special process models are included for effects such as form loss, flow at an abrupt area change, branching, choked flow, boron tracking, and non-condensable gas transport.

The system mathematical models are coupled into an efficient code structure. The code includes extensive input checking capability to help the user discovering input errors and inconsistencies. Also included are free-format input, restart, re-nodalisation, and variable output edit features. These user conveniences were developed in recognition that generally the major cost associated with the use of a system transient code is in the engineering labour and time involved in accumulating system data and developing system models, while the computer cost associated with generation of the final result is usually small. The core of a nuclear power plant can be simulated by several (actually up to a few hundreds) parallel nodes interconnected at different axial elevations by 'cross-junctions' (US NRC, 2001).

1.2. The STHIP-1 Code

The STHIRP-1 computer program (Veloso, 2004) uses the principles of the sub-channels analysis and it is capable to simulate, under steady state and transient conditions, the thermal and hydraulic phenomena occurring inside the core of water cooled research reactor under a natural convection regime. The models and empirical correlations necessary to describe the flow phenomena which can not be described by theoretical relations were selected according to the characteristics of the TRIGA IPR-R1 research reactor operation. The program was validated against the TRIGA-IPR1 model. The calculation results in comparison with the experimental data indicate that the program reproduces the experimental data with good agreements.

1.3. TRIGA IPR-R1 Research Reactor

The RELAP5 code simulation capability has been verified through the simulation of the TRIGA IPR-R1 research reactor behavior. The IPR-R1 is a reactor type TRIGA, Mark-I model, developed by the General Atomic Company. It presents low power (250 kW) 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 high purity water as it is illustrated in the Fig. 1.

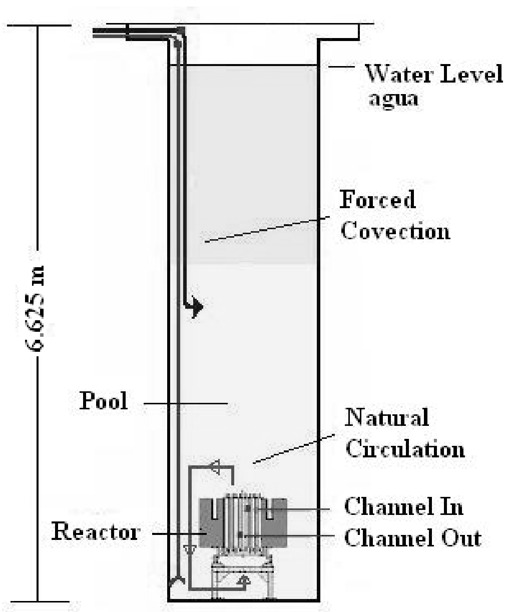

Figure 1. Schematic representation of the TRIGA.

The water in the pool acts mainly as cooling, moderator and neutron reflector. 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. In the forced convection mode, the coolant is forced to flow downward into the core using the pump suction force, while in the natural convection mode the core cooling is maintained by natural convection, flow moving upward through the core.

The core contains 63 fuel elements constituted by a cylindrical metal cladding filled with the fuel. The fuel material is a homogeneous alloy composed by zircon and uranium enriched at 20% in the U^{235} isotope.

2. MODEL DESCRIPTION

Aiming to simulate the TRIGA IPR-R1 research reactor using the RELAP5 code, the reactor pool was modeled using a pipe component divided into ten nodes. 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 components were used and associated with 13 corresponding hydrodynamic pipe components constituting 13 hydrodynamic channels, as can be verified in the Fig. 2. Table 1 presents some characteristics of the 13 regions. Only the natural convection system was modeled. Figure 3 shows the developed layout nodalisation of the TRIGA.

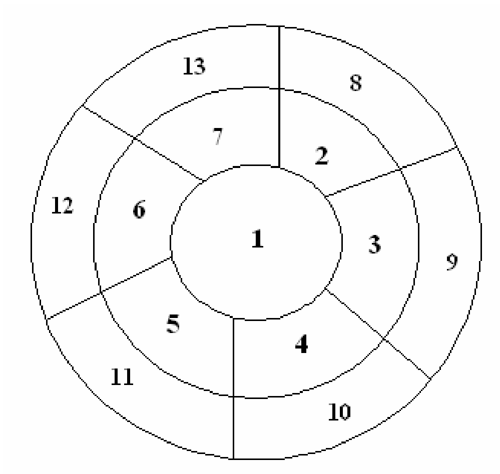

Figure 2. TRIGA planar representation of the 13 thermal hydraulic channels core in RELAP5.

Table 1. Characteristics of the thermal hydraulic regions.

TH Region	Number of Fuel Elements	TH Channel	Sectional Area (m²)	Heat Structures in the TH Channel
1	6	201	0,0110	201 a 206
2	5	202	0,0082	207 a 211
3	5	203	0,0082	212 a 216
4	4	204	0,0082	217 a 220
5	5	205	0,0082	221 a 225
6	5	206	0,0082	226 a 230
7	4	207	0,0082	231 a 234
8	5	208	0,0155	235 a 239
9	5	209	0,0155	240 a 244
10	4	210	0,0155	245 a 248
11	5	211	0,0155	249 a 253
12	5	212	0,0155	254 a 258
13	5	213	0,0155	259 a 263
	Total = 63			

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 location of the fuel assemblies. The radial power distribution shown in the Fig. 4 was computed with WIMSD4C and CITATION codes (Dalle *et al.* 2003).

The axial power distribution was calculated considering a cosine profile. Although the above modeling procedure is approximate, it is used here to maintain the actual axial and radial power distribution fixed.

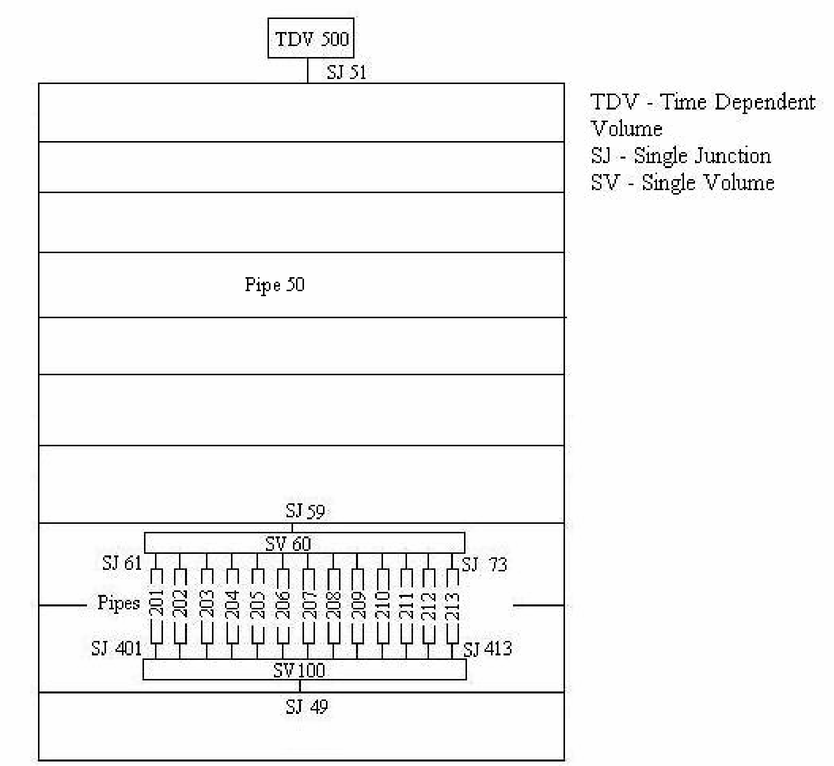

Figure 3. TRIGA nodalisation.

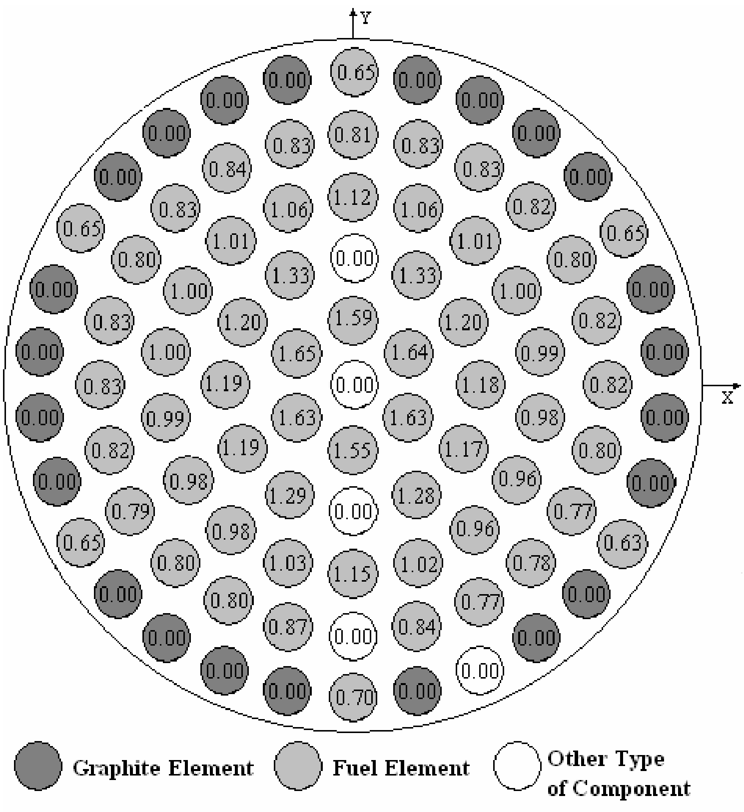

Figure 4. Radial relative power distribution.

3. STEADY STATE CALCULATION

The validation of the RELAP5 nodalisation must to demonstrate that the RELAP model reproduces the measured steady state conditions of the TRIGA IPR-R1 with acceptable margins. The RELAP5 steady state calculation has been performed at two different power levels (50 and 100 kW). The temperature values at the inlet and outlet of the thermal hydraulic channels 3, 8 and 13 calculated by RELAP5 can bee seen in the Tab. 1 and Tab. 2 for 50 kW and 100 kW thermal powers, respectively. The calculated values were compared with the available experimental data (outlet channel temperature) and with to the STHIRP-1 calculation data (Veloso, 2004).

As can be verified in the Tab.1, the results of the calculation are in good agreement with the experimental data. The outlet temperature values calculated by the RELAP5 code are very similar in the three considered channels. The same occurs for the inlet temperature values. The STHIRP-1 code demonstrates to be more sensitive in the determination of the local temperature. It is convenient to remember that the STHIRP-1 code was developed specially according with the TRIGA IPR-R1 characteristics and the core region was modeled with 104 thermal hydraulic channels against 13 thermal hydraulic channels in the present RELAP5 model.

Table 1. Experimental and calculated results for 50 kW thermal power condition.

Power = 50 kW							
Thermal Hydraulic Channel	Outlet Temperature (°C)			Inlet Temperature (°C)		Pressure Drop (kPa)	
	Experimental	RELAP5	STHIRP-1	RELAP5	STHIRP-1	RELAP5	STHIRP-1
3	27	26.0	28.1	27.2	26.1	6.0	8.8
8	25	25.8	25.1	26.5	24.2		
13	25	25.8	26.3	26.5	25.0		

The same behavior is observed at 100 kW power operation, as shown in the Tab. 2. Although both codes to present results in good agreement with the experimental data, the RELAP5 presents constant inlet temperature. In the future, the thermal hydraulic core channels number will be increased in the RELAP5 model to verify again this temperature values. Also the pressure drop in the thermal hydraulic core channels has been calculated by both codes presenting good agreement between them.

Table 2. Experimental and calculated results for 100 kW thermal power condition.

Power = 100 kW							
Thermal Hydraulic Channel	Outlet Temperature (°C)			Inlet Temperature (°C)		Pressure Drop (kPa)	
	Experimental	RELAP5	STHIRP-1	RELAP5	STHIRP-1	RELAP5	STHIRP-1
3	31	30.8	32.2	28.7	29.0	6.0	8.8
8	28	29.7	27.0	28.7	25.4		
13	30	29.8	29.1	28.7	26.9		

4. CONCLUSIONS

The RELAP5 code has been installed in the Nuclear Engineering Department (DEN) of the UFMG and has been validated against the TRIGA IPR-R1 research reactor. In this work, a nodalisation for the TRIGA IPR-R1 research reactor for RELAP5 calculation has been developed as a contribution to the assessment of RELAP5/3.3 for research reactor safety analysis.

The nodalisation was validated against experimental data from steady state conditions. The RELAP5 results have been also compared with some data obtained by the already validated STHIRP-1 thermal hydraulic code. The results showed good agreement between the codes with little discrepancies which could be explained by the different empirical correlations embedded within each code. The little discrepancies can be also related to the differences on the nodalisation methodologies adopted for each code. Future investigations will be performed to verify the action of the number of thermal hydraulic channels in the calculation results. Furthermore, transient calculations will be performed to as a second step in the code validation process.

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.

5. ACKNOWLEDGEMENTS

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