Safety Studies and General Simulations of Research Reactors Using Nuclear Codes

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1. Introduction

Interest in safety issues of nuclear research reactors is nowadays increasing due their enlarged commercial exploitation commonly directed at neutrons generation for several types of scientific and social purposes. Power generation is not the main activity of a nuclear research reactor reaching maximum power operation of about 100 MW. In spite of this, specific features are necessary to ensure safe utilization of such installations. Therefore, several codes have been used focusing special attention for research reactors safety analysis and valuation of specific perturbation plant processes. A combination of codes for thermal hydraulic analysis, for assessment of probabilistic risk, fuel investigation and reactor physics studies are fundamental tools for an appropriate reactor behaviour definition.

It is appropriate to use internationally recognized, accepted and validated best estimate codes. The continuous development and validation of the nuclear codes ensures the improvement of best estimate methods. Typically, thermal hydraulic system codes may need the most effort in terms of developing input models for system analyses in research reactors. The fuel codes can be used for analysis of design basis accident conditions and may be used to provide initial conditions for the system thermal hydraulic codes. Neutron kinetic codes can be coupled to thermal hydraulic system codes to provide a more realistic simulation of transients where there is a large reactivity variation. Reactor physics codes are typically used to support the performance of the core as well as to provide results used in the system thermal hydraulic codes for accident analysis. Containment codes may be necessary to estimate parameters as the time of failure of the containment, confinement or reactor building.

In this Chapter, the state-of-the-art related to nuclear codes applied to research reactors are being presented. Results of simulations performed with two specific codes, the thermal hydraulic RELAP5 code and the General Monte Carlo N-Particle Transport code (MCNP), for the TRIGA IPR-R1 research reactor in Brazil are also presented.

2. Nuclear research reactors in operation

The main activity of the nuclear research reactors is not connected to power generation. However, they are widely used to several activities as to non-destructive materials testing, radioisotopes production, nuclear medicine, research, and many others fields.
The Research Reactor Database (RRDB) of the International Atomic Energy Agency (IAEA) contains administrative, technical and utilization information on over 670 research reactors including critical and sub-critical assemblies in 69 countries and the European Union. Second the RRDB data, nowadays there are 239 research reactors in operation around the world (see Table 1). Approximately, half of this total is now over 40 years old being necessary to address deficiencies and new requirements that evolve over time. In this way, reactor organizations undertake an array of work activities to either re-establish performance that has degraded over time, maintain performance in the face of changing conditions or adapt to new customer or regulatory demands (IAEA, 2009).

Status	**Developed Countries**	**Developing Countries**
Cancelled	1	4
Operational	148	91
Shut down	183	21
Decommissioned	194	16
Temporary shutdown	8	5
Planned	1	1
Under construction	2	1
Unverified information	0	1
Total	**537**	**140**

From IAEA (2011) http://nucleus.iaea.org/RRDB

Table 1. Research reactors in the world

The operating mode as well as the design of research reactors can vary largely differently from the power reactors. The most common design of research reactors is the pool type, where the core is a cluster of fuel elements sitting in a large pool of water. The water in the pool has function of cooling, as well as moderation, neutron reflector and it is 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. The heat generated from the nuclear fissions can be also removed pumping the pool water through a heat exchanger characterizing a forced cooling.
Some examples of different types of research reactors are listed in Table 2. The TRIGA reactor is the most common design having about 60-100 cylindrical fuel elements with metal cladding enclosing a mixture of uranium fuel and zirconium hydride. The main characteristic of this type of fuel is the prompt negative temperature coefficient that provides safety and automatically limiting the power when excess of reactivity is suddenly inserted. Fig. 1 shows a photography of an upper view of the research reactor type open-pool IPR-R1 TRIGA. IPR-R1 is installed at Nuclear Energy Development Centre (CDTN) of Brazilian Nuclear Energy Commission (CNEN), in Belo Horizonte, Brazil. It works at 100 kW but will be briefly licensed to operate at 250 kW. 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.

Other research reactor designs are moderated using heavy water or graphite. The fast reactors are in a small number; they require no moderator and can use a mixture of uranium and plutonium as fuel. Homogenous type reactors have a core comprising a solution of uranium salts as a liquid, contained in a tank about 300 mm diameter. This type was popular in the past due to its simple design; however only a small number is nowadays in operation. High temperature research reactors, as that developed in Japan (the HTTR – High Temperature Test Reactor), have mainly the aim of to investigate the TRISO fuel designed for the Generation IV power reactors, as the HTGRs - High Temperature Gas Reactors - Verfondern et al., 2007).

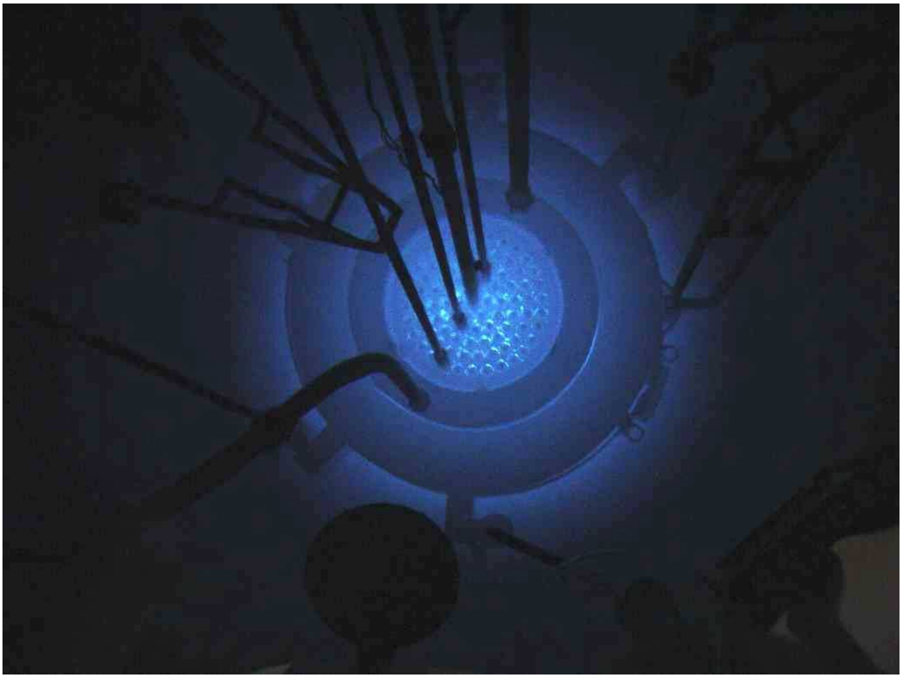

Fig. 1. Core upper view and pool of the IPR-R1 TRIGA

Graphite or beryllium is commonly used as the reflector in research reactors, although other materials may also be used. The fuel of research reactors can be of type HEU (highly enriched uranium) or LEU (low enriched uranium). However, because of the programmes of nuclear non-proliferation, there is a tendency of the countries to convert core reactor HEU to LEU.

The fuel assemblies of research reactors are typically made in plates or cylinders, as presented in the examples of Figure 2 and Figure 3, respectively. Figure 2 illustrates the MTR (material testing reactor) fuel assembly used in the IEA-R1. A typical IEA-R1 fuel element has 18 plane parallel fuel plates, mounted mechanically between two lateral aluminium holders with grooves, and its overall dimensions are (7.6 X 8.0) cm and 88.0 cm high. Each fuel plate consists of an aluminium cladding and a meat where the nuclear fuel is located (Terremoto et al., 2000).

Figure 3 presents the design of two types of cylindrical fuel elements used in the research reactor IPR-R1. This fuel contains high concentrations of hydrogen using a metal alloy of uranium and zirconium and its main characteristic is the prompt negative temperature coefficient.

Type	Name	Country	Power (kW)	Criticality	Thermal Flux (n/cm²s)	Fuel and Enrichment
TRIGA	IPR-R1	Brazil	100	1960	4.3×10^{12}	U-Zr-H 20%
Pool	IEA-R1	Brazil	5000	1957	4.6×10^{13}	U_3O_8-Al and U_3Si_2-Al 20%
Pool MTR	MNR	Canada	5000	1959	1.0×10^{14}	U_3Si_2-Al 19.75%
Fast Source	TAPIRO	Italy	5	1971	(Fast Flux) 4.0×10^{12}	U-Mo alloy 93.5%
High Temperature Gas	HTTR	Japan	30000	1998	7.5×10^{13}	UO_2 6%
Argonaut	UFTR	USA	100	1959	2.0×10^{12}	U_3Si_2-Al 19.75%
Heavy Water	NBSR	USA	20000	1967	4.0×10^{14}	U_3O_8-Al 93%

From IAEA (2011) http://nucleus.iaea.org/RRDB

Table 2. Examples of nuclear research reactors

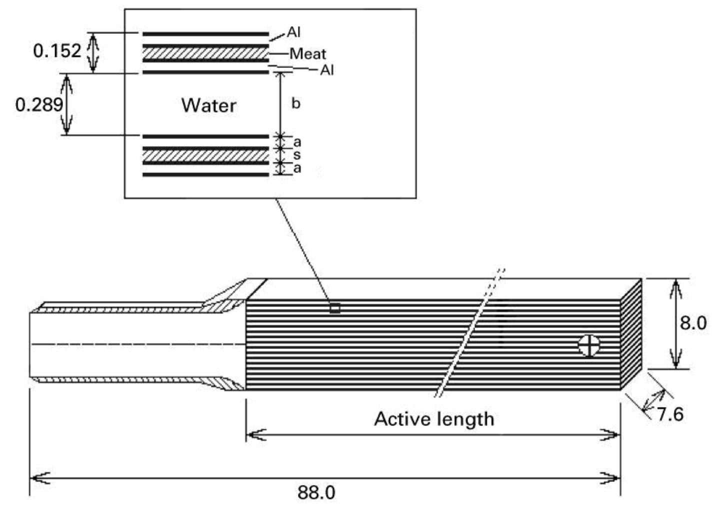

Fig. 2. Cross-sectional diagram of a standard MTR fuel element irradiated in the IEA-R1 research reactor, showing in detail the structure of two successive fuel plates (measure in cm). Adapted from (Terremoto et al., 2000)

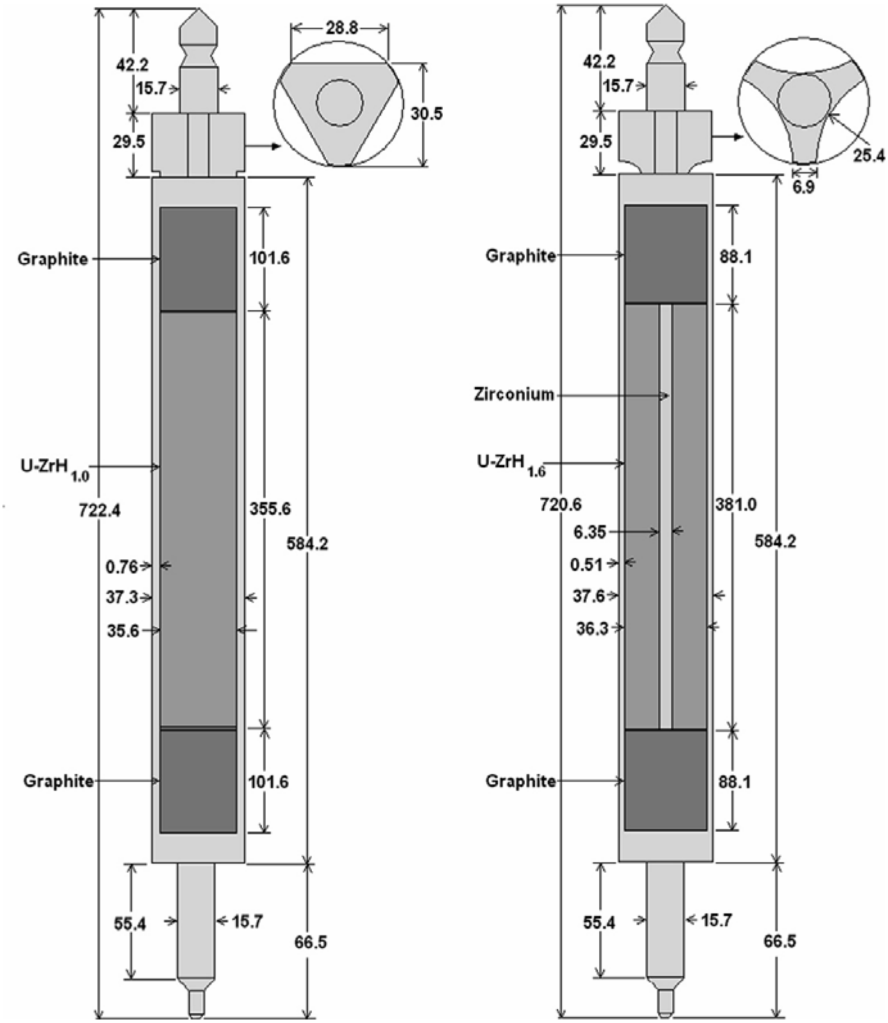

Fig. 3. IPR-R1 TRIGA – design of two types of cylindrical fuel elements (measure in mm)

3. Application of nuclear codes in research reactor analysis

In general, the codes used for research reactors analysis are also used in the nuclear power plant (NPP) having both the same basis of development and utilization. The differences on validation and application for each case appear due the complexity of the different classes of reactors. Particularly, the codes available internationally for safety analysis of research reactors can be classified in different issues according with their application including reactor physics, fuel behaviour, thermal hydraulic processing, computational fluid dynamics (CFD) and structural analysis (IAEA, 2008). Each of these topics are being explained with some more details and exemplified next.

3.1 Neutron kinetic modeling

Reactor physics codes are capable to model the 2D or 3D core neutron kinetics for analysing local or asymmetrical effects in the reactor core that is possible to occur as in steady state as in transient operation. Examples of reactor physics codes are WIMS-D, DYN3D, CITATION, PARET and NESTLE. WIMS-D, a deterministic code system for reactor lattice calculation, can be used to calculate group constants dividing the core into several identical unit cells. The calculated cross-sections can then be used as input to another type of code as, for example, the code CITATION for global core calculations (Khana et al., 2000; Dalle et al., 2002).

The computer code DYN3D was developed for safety analyses of nuclear reactors after reactivity perturbations of the system, but it can be used also for fuel management calculations.

PARET code has been used extensively for research reactor analysis which iteratively solves for the neutronic-hydrodynamic-heat transfer aspect of the reactor under steady state and transient behaviour. It can be used to investigate core reactivity insertions being a coupled kinetics and thermal hydraulics code for predicting the course of non-destructive transients in research reactors (Woodruff et al., 1996; Housiadas, 2002; Velit and Primm, 2008).

Other example is the NESTLE code that solves the two or four group neutron diffusion equations in either Cartesian or hexagonal geometry using the Nodal Expansion Method (NEM) and the non-linear iteration technique. NESTLE was embedded in the thermal hydraulic code RELAP5 obtaining the multi-dimensional neutron kinetics model RELAP5-3D. Steady-state eigenvalue and time dependent neutron flux problems can be solved by the NESTLE code as implemented in RELAP5-3D. In spite of RELAP5-3D to be developed for power reactors applications, it has been successfully used for research reactors analyses (Costa et al., 2011; Marcum et al., 2010).

Therefore, as can be verified from some before examples, generally two or more codes are used directly or indirectly connected for a more detailed and realistic simulation exploring the main capability of each one.

Calculations using discrete ordinate diffusion and transport theory have been used extensively for reactor simulation purposes. However, the Monte Carlo technique offers significant advantages, since the complex geometrical configuration of the reactor core can be modelled in detail. Therefore, the Monte Carlo code (MCNP) has been applied to research reactor simulations mainly for neutron flux calculations (Fernandes et al., 2010; Shoushtari et al., 2009; Stamatelatos et al., 2007; Huda, 2006). A more detailed example of the MCNP code application to simulations of neutron flux value on the irradiation channels of the IPR-R1 TRIGA research reactor, adapted from (Guerra et al., 2011), has been presented in the Annex A.

3.2 Fuel analysis

Researchers in several countries have worked with the aiming to develop codes that predict the behaviour of a fuel assembly during extreme transients as, for example, a LOCA (loss of coolant accident). Such codes attempt to predict the deformation of a fuel rod, the termination of deformation by rupture, the temperature reached by the cladding, oxidation of cladding, and in some codes, the interaction between neighbouring rods. Codes which calculate fuel rod behaviour in whole assemblies including rod-to-rod interactions are relatively rare. One example of such code is the Japanese FRETA-B specialised in two-dimensional analysis in the transverse direction (NEA, 2009). DRACCAR is other example of fuel code that is currently under development at IRSN (Institute for Radiological Protection and Nuclear Safety) with the purpose of to simulate the thermal mechanical behaviour of a rod bundle under LOCA with a 3D multi-rod description (Papin et al., 2006).

3.3 Thermal hydraulic modeling

Thermal hydraulic system codes are applicable to a wide variety of reactor designs and conditions. Such system codes allow simulating the complete primary and secondary circuits and the interactions between them. Examples of system thermal hydraulic codes are RELAP5, TRAC, CATHARE, ATHELET, DINAMIKA and CATHENA. They are generally classified as best estimate codes. The term "best estimate code" means that the code is free of deliberate pessimism and contains sufficiently detailed models to describe the relevant processes of the transients that the code is designed to model (IAEA, 2008).

Models for two fluid, non-equilibrium hydrodynamics, point and multidimensional reactor kinetics, control systems, and special system components make these thermal hydraulic codes very attractive. However, the use of these codes for research reactors must be careful in order to ensure that the models included in such codes are valid for the operating regimes of the research reactors. The validity of the models and correlations should be verified.

As an example, the ATHLET thermal hydraulic code developed at the GRS, Society for Plant and Reactor Safety, was planed to analyse leaks and transients for power reactors. However, to extend the applicability of the code to the safety analysis of research reactors, a model was implemented permitting a description of the thermal-dynamic non-equilibrium effects in the subcooled boiling regime (Hainoun et al., 1996).

In the same way, the RELAP5 code has been modified to better simulate the research reactors operation conditions (low pressure, low mass flow rate, low power). For example, a subcooled boiling model of upward vertical flow consistent with phenomenological observations of the subcooled flow boiling mechanisms was proposed to extend the range of applicability of the RELAP5 code to low pressures (Končar and Mavko, 2003). Therefore, recent works as, for example (Antariksawan et al., 2005; Khedr et al., 2005; Marcum et al., 2010; Reis et al., 2010), have been performed to investigate the applicability of the RELAP5 code to research reactors operating conditions (TRIGA 2000, MTR, Oregon State TRIGA, IPR-R1 TRIGA), respectively. Application of a model for the IPR-R1 TRIGA using the RELAP5 code is detailed in the Annex B.

The user of a thermal hydraulic system code has a very large number of available basic elements (single volumes, pipes, branches, junctions, heat structures, pumps, etc) to develop a detailed reactor nodalization. The model can reproduce a specific part or the whole system to be simulated. However as there is not a fixed rule to perform the nodalization, a large responsibility is passed to the user of the code in order to develop an adequate model scheme which makes best use of the various modules and the prediction capabilities of the specific code (Petruzzi and D'Auria, 2008; D'Auria and Galassi, 1998).

Subchannel codes are used to analyse specific processes within the core of the reactor, such as localized flow and heat transfer variables in representative fuel assemblies. Examples are PARET and COBRA codes.

Computational Fluid Dynamics (CFD) is increasingly being used in the nuclear community to model safety relevant phenomena occurring in the reactor coolant system and for the analysis of localized phenomena such as the flow pattern in complex geometries. However, CFD is a relatively recent development and their qualification status for application in transient flow analysis for research reactor licensing should be verified.

3.4 Structural codes

Structural analysis codes are used to describe the behaviour of mechanical components such as core support and pool structures, in the case of a pool type reactor, under various accident conditions. These codes are commercially available and have generally been developed for non-nuclear applications. They utilize boundary conditions supplied, for example, by thermal hydraulic codes. Examples of structural analysis codes are NASTRAN and ANSYS.

NASTRAN, the NASA Structural Analysis System, is a powerful general purpose finite element analysis program and it is a standard in the structural analysis field, providing the engineer with a wide range of modelling and analysis capabilities. The computational programme ANSYS is a multipurpose finite element code that can perform a variety of calculations, including stress analysis, temperature distributions, and thermal expansions in solid materials.

4. Verification and validation of codes

The applicability of a code to reactor safety analysis, mainly for licensing, is directly related with its qualification which must be rigorously documented. It is not possible to provide a detailed list of the key phenomena and code features necessary for each type of code. However, the IAEA proposes basically three criteria to verify the adequacy of the codes for treating important phenomena (IAEA, 2008):

a. The use of internationally recognized and accepted codes provides some assurance that the codes are adequate for their intended application.
b. Individual codes need to be evaluated on a systematic basis, comparing the intended application of the code with the actual conditions for which the code is applied.
c. Lists of important phenomena expected during the transients that constitute the target of the investigation must be established. In many cases, documentation is available on an individual code basis that describes the relative importance of the different phenomena.

Code verification is defined as the review of the source coding against its description in the documentation. The line by line verification of large codes is a time consuming and expensive process. Therefore, this process is limited to only some codes. However, many industry sponsored codes have been subjected to stringent verification procedures as a consequence of the regulatory licensing process (IAEA, 2008).

Extensive code validation requires efforts at the international level, involving validation projects, usually managed by the code developers and carried out, under cooperation and exchange agreements, by user groups worldwide with access to experimental facilities designed to provide data on behaviour and phenomena of importance. Several international standard problems provide comparison between codes.

The validation of a code modelling 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 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.

Sensitivity analysis including systematic variations in code input variables or modelling 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., 2011).

5. Safety analysis criteria

The acceptance criteria are essential to classify the results obtained from a safety analysis and they may be specified as basic and specific. Basic acceptance criteria are usually defined as limits set by a regulatory body. The specific acceptance criteria are used to include additional margins beyond the basic acceptance criteria to allow for uncertainties and to provide additional defence in depth. The margin between results predicted by the analysis and the acceptance criterion is related to the uncertainties. If a result has low uncertainty, a small margin to the acceptance criteria may be acceptable. In general, the adequacy of the margin with the acceptance criterion is demonstrated by using a conservative analysis to meet the acceptance criterion (IAEA, 2008).
Deterministic techniques are the main tools used in the analyses of research reactors. These techniques are often related with the conservatism, commonly knew as conservative approach. On the other hand, 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. A best estimate analysis must be supplemented by an uncertainty analysis. Application of best-estimate (realistic) computer codes to the safety analysis of nuclear plants implies the evaluation of uncertainties. This is connected with the (imperfect) nature of the codes and of the process of codes application. The source of uncertainties affects the predictions by best-estimate codes and must be taken into account (D'Auria, 2004).

6. Reactor parameters in the safety analysis

The safety analyses are used in several areas including design, licensing, support for accident management and emergency planning. Reactor parameters and some operating conditions considered in the safety analysis can be summarized in: state of the reactor operation, core power, core inlet temperature, fuel element cladding temperature; system pressure, core flow, axial and radial power distribution and hot channel factor; reactor kinetics parameters, fuel and moderator temperature reactivity coefficients, void reactivity coefficient, available shutdown reactivity worth and insertion characteristics of reactivity control and safety devices.
The evaluation of the safety of research reactors includes firstly the determination of the reactor response to a range of postulated initiating events (PIEs) covering all supposed possible types of events. Several selected PIEs for research reactors have been classified and summarized as follows (IAEA, 2005):

a. Loss of electric power supplies;
b. Insertion of excess reactivity;
c. Loss of flow;
d. Loss of coolant;
e. Erroneous handling or failure of equipment;
f. Special internal events;
g. External events;
h. Human errors.

The PIEs in each group should be evaluated to identify the events that would be limiting, and from there the events selected for further analysis should be indicated. Such events would include those having potential consequences that bound all other PIEs in the group.
For example, to 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 reactor. In spite of this, some perturbation situation may occur disturbing the normal reactor operation, as a condition of forced recirculation off. The event can be caused by the recirculation pump failure e can be classified inside the event number 3 described before. Annex B presents an example of simulation of this event using a TRIGA model in the RELAP5 code with results very approximate from the experimental data.

7. Conclusion

This Chapter has drawn the attention to specific features related to the safe utilization of research reactors. A summarized state-of-the-art about research reactors was presented.
As it was illustrated, the rising interest in the commercial exploitation of these types of nuclear reactors is justified by the several applications using mainly their neutrons generation. Due the considerable age of the majority of research reactors in operation around the world, it is necessary to address deficiencies and new requirements that evolve over time. Works have been performed with the aim of either re-establish performance that has degraded over time, maintain performance in the face of changing conditions or adapt to new customer or regulatory demands. As part of these efforts, several codes have been used focusing special attention for the research reactors safety analysis including thermal hydraulic analysis, fuel investigation, reactor physics and structural studies. Codes internationally recognized, accepted and validated are essential in reactor safety analysis that are used in several areas including design, licensing, support for accident management and emergency planning.
An important aspect in the applicability of a code to reactor safety analysis is its direct relation with its qualification which must be rigorously documented. Moreover aspects connected to the validation, uncertainty analysis and sensitivity analysis must be carefully considered for a correct application of a nuclear code for the simulation of reactor model.
Examples of application of two types of codes widely used for research reactors analysis are being presented in the Annex A and B (MCNP and RELAP5 codes, respectively). These Annexes present results of simulations performed at the Nuclear Engineering Department of the Federal University of Minas Gerais for the TRIGA IPR-R1 research reactor in Brazil.

8. Acknowledgment

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ANNEX A. Example of MCNP code application to IPR-R1 research reactor

The Annex A and also the Annex B are based in the IPR-R1 research reactor and they are a summary of the works (Guerra et al., 2011) and (Reis et al., 2010), respectively.

AI. Introduction

The IPR-R1 is a reactor type TRIGA, Mark-I model that is installed at Nuclear Technology 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. IPR-R1 works at 100 kW but it will be briefly licensed to operate at 250 kW. It presents low power, low pressure, for application in research, training and radioisotopes production. The fuel is an alloy of zirconium hydride and uranium enriched at 20% in ^{235}U.

The IPR-R1 reactor has a Rotary Specimen Rack, RSR, outside the reactor, and it is composed by forty irradiation channels in a cylindrical geometry. Moreover, tangent to annular reflector, there is a Pneumatic Tube where the samples also can be inserted to irradiation. Therefore, the IPR-R1 has three facilities for sample irradiation: the Central Thimble, the Rotary Specimen Rack and the Pneumatic Tube. Figure A1 shows the radial and axial core configuration. The IPR-R1 has the main nuclear applications related with neutron activation analysis (NAA), training of operators for nuclear power plants and experiments in neutronic and thermal hydraulic.

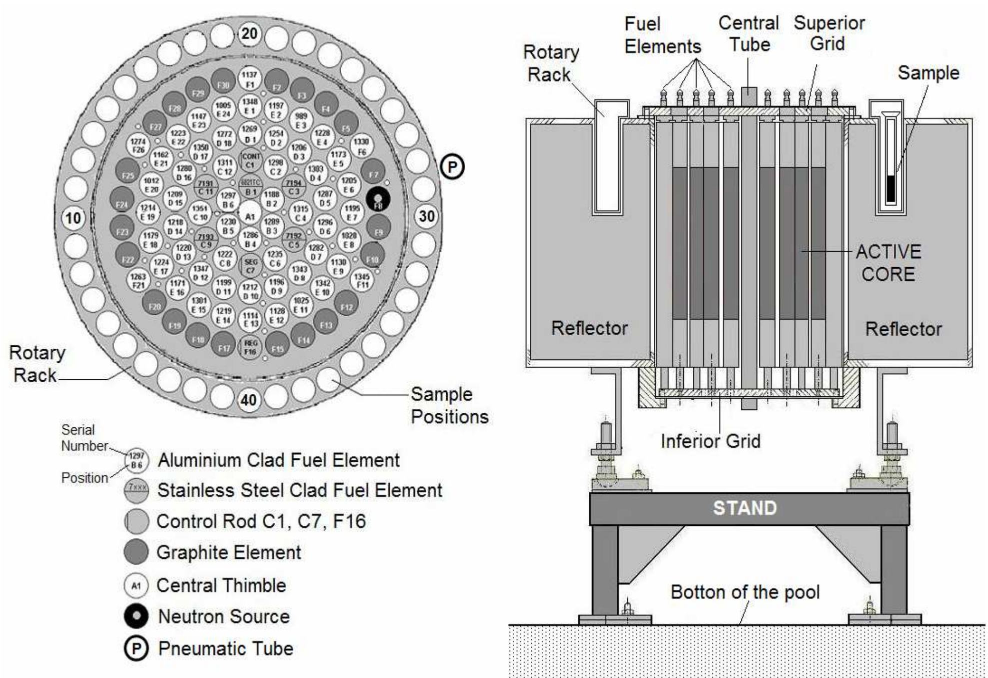

Fig. A1. Core upper view and pool of the IPR-R1 TRIGA

The IPR-R1 has been simulated using the code MCNPX2.6.0 (Monte Carlo N-Particle Transport eXtend). The goal was to evaluate the neutronic flux in a sample inserted in the RSR channels. In each simulation the sample was placed in a different position, totalling forty positions around the RSR. The results obtained from the calculation show good agreement in relation to experimental data.

AII. Modelling

The reactor model was developed using MCNP5 and MCNPX codes where 500 active cycles were calculated with 1000 neutrons per cycle. The simulations executed in MCNPX code, considers 1.0 hour of irradiation (time of sample irradiation) with 0.1 MW (currently TRIGA reactor power). The simulated model was based on previous studies where the reactor core has the same features of the IPR-R1 geometry described before. The configured geometry is the same to MCNP5 and the MCNPX 2.6.0. The core was configured considering a cylinder containing water, fuel elements, radial reflectors, central tube (or central thimble), control rods and neutron source. Each rod has a coordinate value. They were filled according to their individual characteristics. Around the core there is the RSR which has groove to insert the samples to irradiation. The configured core is inside the pool where water surrounds the core and the RSR. However, the model of this work presents some improvements.
Table A1 presents the cylinders dimensions which are inside of RSR. In addition, Figure A2 illustrates the axial and radial view of the simulated model.

Cylinder Material	Inner Radius (cm)	Radial Thickness (cm)	Height (cm)
Aluminum	1.50	0.10	20.0
Polystyrene	1.10	0.30	7.90
Polyethylene	0.48	0.07	0.55

Table A1. Dimensions of the simulated cylinders inside Rotary Specimen Rack

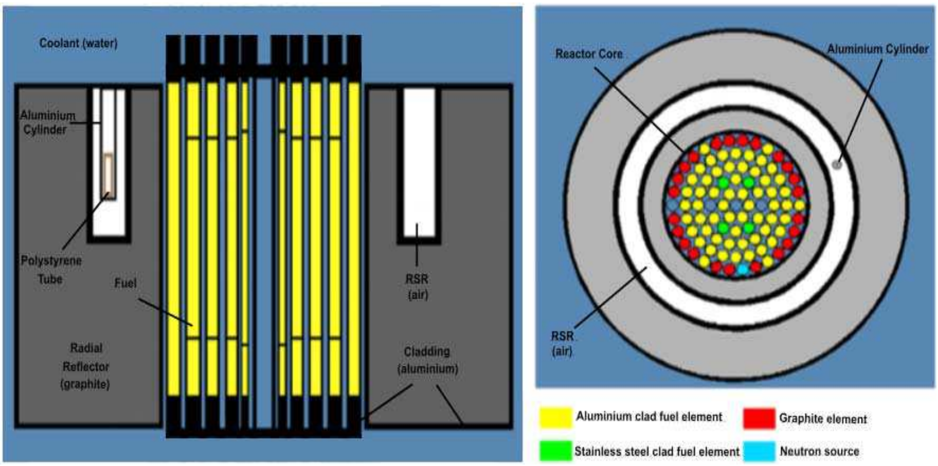

Fig. A2. Axial and radial view of the reactor TRIGA IPR-R1 simulated in MCNP5 and MCNPX codes

AIII. Results and conclusions

The assembly of the three cylinders was positioned in 40 different positions of the RSR to calculate the thermal neutron flux (energies of 0.5 eV or smaller) inside of the polyethylene cylinder that contains the sample. The Figure A3 shows the thermal neutron flux simulated by MCNP5 and MCNPX codes. It is possible to see clearly the behaviour of the thermal neutron flux around the core reactor. The neutron fluxes vary at each position and changes over the RSR. In spite of the good agreement of the most of the calculated points some differences were observed. These differences must be justified by a theoretical analysis and using a statistical program.

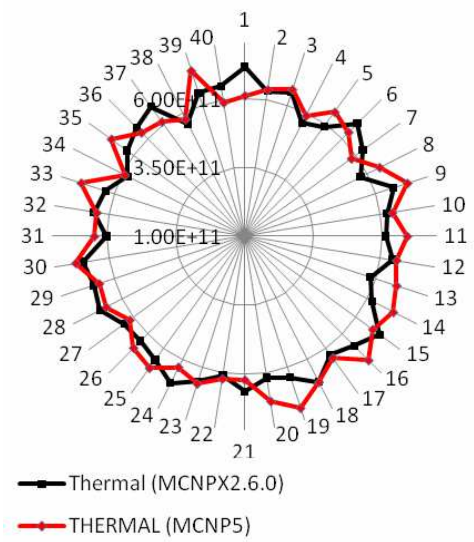

Fig. A3. Axial and radial view of the reactor TRIGA IPR-R1 simulated in MCNP5 and MCNPX codes

Although the neutron flux value was calculated in 40 positions in the RSR, the reference work presents only 11 experimental data. Table A2 presents the values of thermal neutron flux calculated by the codes MCNP-4B, MCNP5 and MCNPX 2.6.0. The results are being compared with experimental data and the error is presented in percentage.

Eight positions present differences smaller than 10% and three positions present differences smaller than 21%. In this way, it is possible to conclude that the most of the calculated results are in good agreement with the experimental measurements.

The results showed that there is good agreement between the experimental data and the values calculated by the used codes. The errors presented by the most of estimated flux are smaller than 10% and three positions presented differences between 10 and 21%. The tool employed by MCNP5 and MCNPX 2.6.0 estimate the flux average inside of the cylinder which contains the sample. However, the used codes have other method to calculate the neutron flux as Forced Collision, Point Detectors, Spherical Detector, etc. These tools may improve the results decreasing the differences between experimental data and the calculated values. For more details of this work see (Guerra et al., 2011).

Future woks will simulate the IPR-R1 employing other method to flux calculate. The information about neutron flux predicted by MNCP5 and MCNPX 2.6.0 can improve NAA where the sample activity can be estimated knowing neutron flux. Furthermore, these codes can characterize the neutron flux in other parts of the reactor where experimental measuring is difficult to be obtained.

Position RSR	Previous Studies			Present Study			
	Experi-mental Value	Model 1 (MCNP 4B)		Model 2 (MCNP5)		Model 3 (MCNPX)	
		Value	Error	Value	Error	Value	Error
1	6.69	6.77	1.18	6.11	8.67	7.14	6.30
3	6.55	6.65	1.50	6.60	0.76	6.50	0.76
7	6.35	6.67	4.80	5.79	8.82	6.32	0.47
10	5.99	6.90	13.19	6.44	6.99	6.24	4.01
24	6.94	6.98	0.57	6.33	8.79	6.97	0.43
25	6.45	6.86	5.98	6.91	6.66	6.54	1.38
29	7.32	6.86	6.28	6.57	10.25	6.77	7.51
34	7.30	6.73	7.81	5.90	19.18	5.77	20.96
35	7.18	6.72	6.41	7.00	2.51	6.29	12.40
38	6.58	6.80	3.24	5.76	12.46	5.58	15.20
40	6.16	6.73	8.47	5.91	4.06	6.51	5.38

Table A2. Thermal neutron flux (x 10^{11} n/cm^{-2}s^{-1})

ANNEX B. Example of RELAP5 code application to IPR-R1 research reactor

BI. Introduction

The RELAP5 system code was developed to simulate transient scenarios in power reactors such as PWR and BWR but recent works have been performed to investigate the applicability of the code to research reactors operating conditions with good results.
Specifically, the TRIGA reactors are constructed in a variety of configurations and capabilities, with steady-state power levels ranging from 20 kilowatts to 16 megawatts offering true "inherent safety". TRIGA is a pool-type reactor that can be installed without a containment building being designed for use by scientific institutions and universities for purposes such as graduate education, private commercial research, non-destructive testing and isotope production.
In the present work, the IPR-R1 TRIGA reactor, Mark-I model, installed in Brazil, in operation since 1960, has been modeled for RELAP5 code with the aim of to reproduce the measured steady-state as well as transient conditions. The development and the calculation for the validation of a RELAP5 model for the IPR-R1 TRIGA research reactor have been presented. The version MOD3.3 was used to perform the simulations. The current results obtained with the developed nodalization demonstrate that the IPR-R1 TRIGA model is representative of the reactor behaviour considering steady-state and transient operation conditions as it is being described in the next sections.

IPR-R1 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. A schematic reactor diagram is illustrated in the Figure B1.

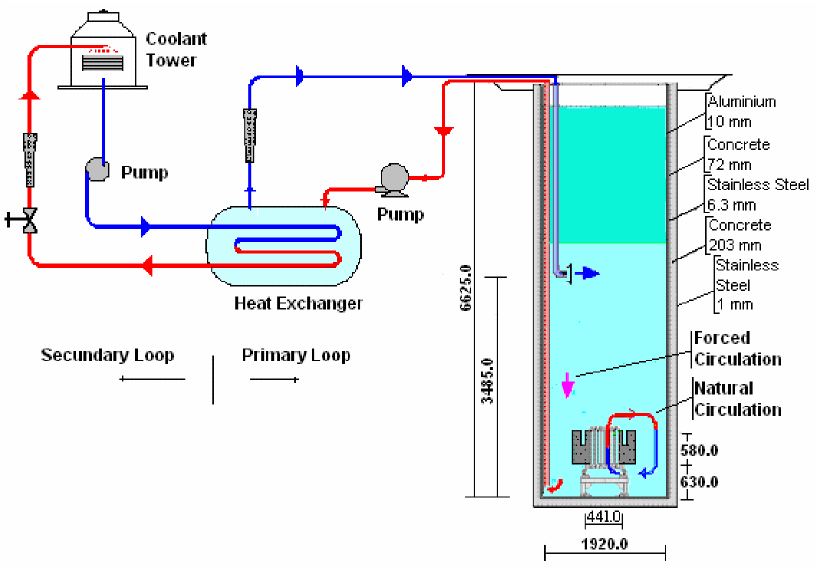

Fig. B1. Schematic representation of the IPR-R1 (out of scale, measure in mm)

The main aim of the water in the pool is for cooling, as well as moderator, neutron reflector and it is 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. The heat removal 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. There are 59 fuel elements covered with aluminium and 4 fuel elements with stainless steel.

BII. Modelling

Each of the 63 fuel elements was modelled 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 Figure B2.
Figure B3 shows the RELAP5 general nodalization developed to simulate the IPR-R1. The reactor pool was modelled using two pipe components, each one composed by ten volumes. As it can be verified by the Figure B3, both components (020 and 050) have their volumes connected by single junctions to characterize a cross flow model. This model improves transient predictions as it will be clearly demonstrated in the transient results. A time

dependent volume was used to simulate the atmospheric pressure on the pool surface. The natural convection system and the primary loop circulation have been modelled. The secondary loop, composed mainly by the external cooling tower was not modelled in the present nodalization because the primary circuit was sufficient to guaranty the heat removal of the coolant.

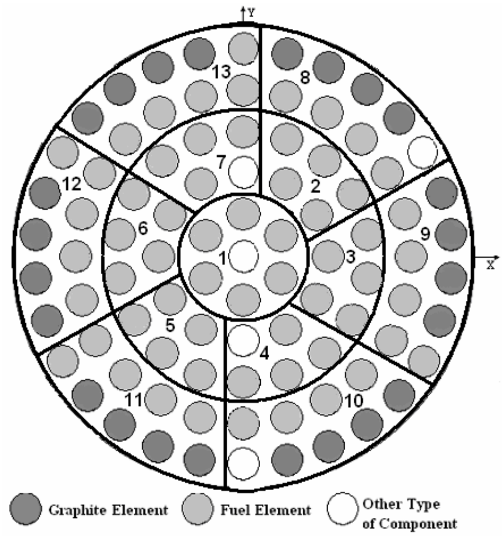

Fig. B2. Representation of the 13 TH channels in RELAP5 model

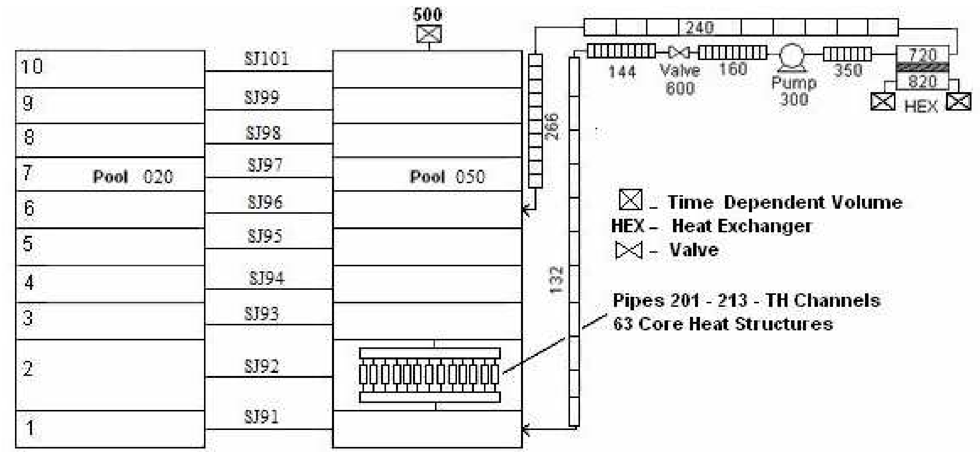

Fig. B3. IPR-R1 TRIGA nodalization in the RELAP5 model

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 also that the power is cut off in the extremes of the element due the presence of the graphite as it is sketched in the Figure B4. Although the above modelling procedure is approximated, it is used here to maintain the actual axial and radial power distribution fixed.

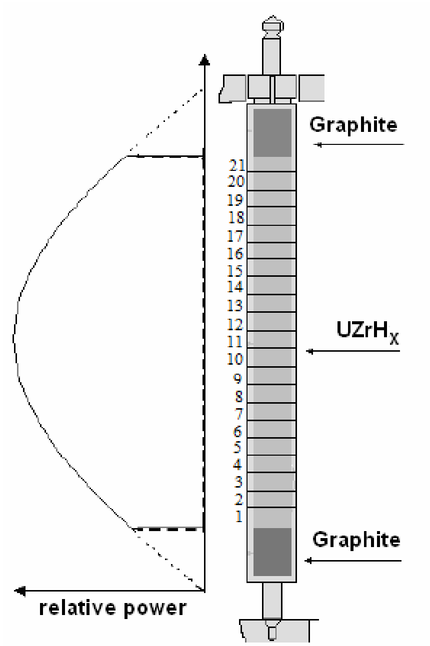

Fig. B4. Prediction of the axial power distribution function in a TRIGA fuel element

BIII. Steady state results

The validation of a RELAP5 nodalization 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 satisfactory time evolution conditions. The RELAP5 steady state calculation has been performed at 50 and 100 kW. The temperature values at the inlet and outlet of the thermal hydraulic channels 3, 8 and 13 calculated using RELAP5 can be verified in the Tables B1 and B2, for 50 e 100 kW, respectively. The calculated values were compared with the available experimental data (inlet and outlet channel temperature). Chromel-alumel calibrated thermocouples were used to collect the coolant temperature data and the measured values have a maximum error of ±1°C.

As it can be verified in the Table B1, considering operation at 50 kW, the results of the RELAP5 code are in good agreement with the experimental data. The error obtained using the RELAP5 calculation is into the range of the maximum acceptable error suggested for coolant temperature (0.5 %) by the RELAP5 users.

TH Channel	Outlet Channel Temperature (K)			Inlet Temperature (K)		
	Experi-mental	RELAP5	Error (%)*	Experi-mental	RELAP5	Error (%)*
3	300.0	298.4	0.5	294.1	294.7	0.1
8	298.0	296.4	0.5	296.1	294.7	0.5
13	298.0	296.4	0.5	0.4	294.7	0.5

* error = 100 X (Calculation – Experimental)/Experimental

Table B1. Experimental and calculated results at 50 kW of power operation

Results performed at 100 kW of power operation are shown in Table B2. The error found for RELAP5 calculation is a few overestimated in comparison with the error suggested for coolant temperature (0.5 %) by the RELAP5 users. However, considering the error from the experimental data (±1°C) the values predicted using RELAP5 are perfectly acceptable for the present model validation process for operation power up to 100 kW.

TH Channel	Outlet Channel Temperature (K)			Inlet Temperature (K)		
	Experi-mental	RELAP5	Error (%)*	Experi-mental	RELAP5	Error (%)*
3	304.0	301.3	0.9	294.0	295.7	0.6
8	300.5	298.8	0.8	295.5	295.7	0.1
13	301.5	298.8	1.1	296.5	295.7	0.3

* error = 100 X (Calculation – Experimental)/Experimental

Table B2. Experimental and calculated results at 100 kW of power operation

Figures B5 and B6 show the RELAP5 calculation for the inlet and outlet temperature for the TH channel 1, at 50 and 100 kW of power, respectively. Such channel was chosen because it concentrates the HS with higher values of radial power. As it can be verified, after about 2500 s of calculation, the temperatures reach steady-state condition. The temperature stable values are in good agreement with the experimental available data.

BIV. Transient results

In spite of the IPR-R1 to be inherently safe, situations that can disturb the normal reactor operation are possible to occur. The RELAP5 model presented in this work has demonstrated to reproduce very well the steady-state conditions. Therefore, in addition to the validation of the modelling process, a transient event was investigated using the code and the results has been compared with available experimental data. The investigated event is the forced recirculation off and may be caused by the recirculation pump failure, bringing the reactor to operate in natural circulation conditions.

In the experiment, 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 (Mesquita et al., 2009). The measurements have demonstrated an average temperature-rise rate of about 4.8°C/h. At inlet and outlet of a thermal hydraulic channel the temperature values were verified to increase about 5.3 °C/h in both cases.

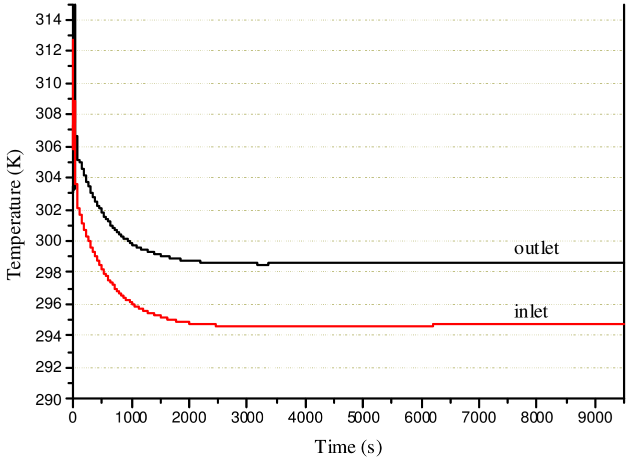

Fig. B5. Inlet and outlet coolant temperature for the channel 1 at 50 kW predicted by the RELAP5

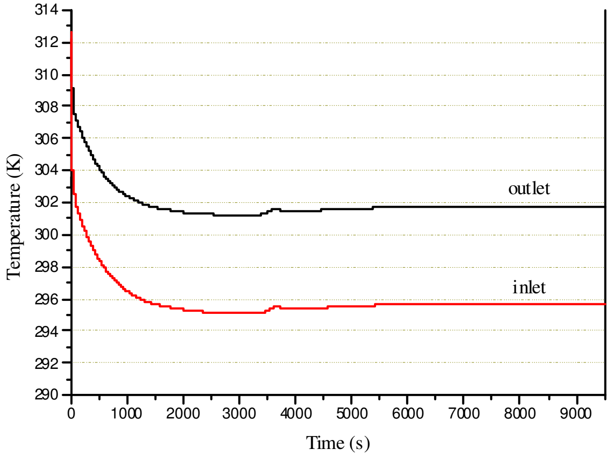

Fig. B6. Inlet and outlet coolant temperature for the channel 1 at 100 kW predicted by the RELAP5

To perform the simulation using the RELAP5, the valve in the primary system (number 600 in the nodalization) has been closed at 3000 s of calculation after the system to reach steady-state condition. After the beginning of the transient, the temperatures increase as consequence of no energy removal from the pool since the primary was off (see Figure B7). After the beginning of the transient, the coolant temperature at inlet and outlet TH channel 1 increased gradually with rates of about 4.9°C/h and 4.6°C/h, respectively, demonstrating very good agreement with the experimental available data.
The insertion of the cross flow model in the pool nodalization makes possible better removal of heat from the core during natural circulation condition due improvement on the coolant

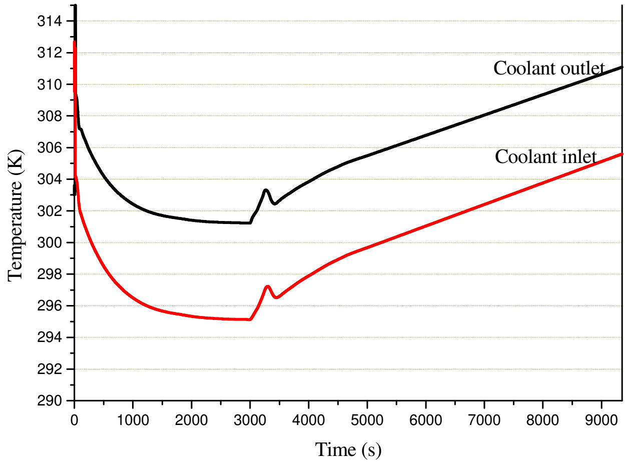

Fig. B7. Inlet and outlet coolant temperature for the channel 1 at 100 kW predicted by the RELAP5 after forced recirculation off at 3000 s

flow between the pool pipe volumes. Figure B8 illustrates the coolant temperature code prediction considering the nodalization presented in this paper and that in the nodalization without cross flow model, both at 100 kW of power operation. The curves show clearly that the model using cross flow presents a temperature-rise rate (4.9°C/h) much more approximated to the experimental (4.8°C/h) than that without cross flow model (30.0°C/h).

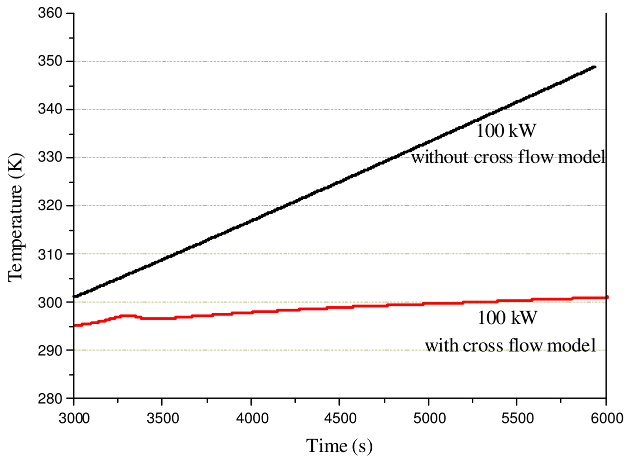

Fig. B8. Forced recirculation off transient prediction using two types of pool nodalization

BV. Conclusion

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 represent adequately the IPR-R1 TRIGA research reactor in steady-state as well as in transient situations.

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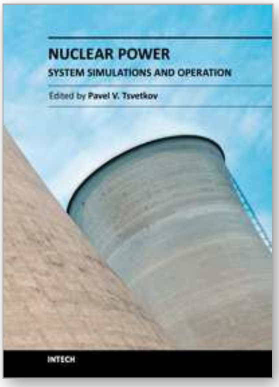

Nuclear Power - System Simulations and Operation
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At the onset of the 21st century, we are searching for reliable and sustainable energy sources that have a potential to support growing economies developing at accelerated growth rates, technology advances improving quality of life and becoming available to larger and larger populations. The quest for robust sustainable energy supplies meeting the above constraints leads us to the nuclear power technology. Todayâ€™s nuclear reactors are safe and highly efficient energy systems that offer electricity and a multitude of co-generation energy products ranging from potable water to heat for industrial applications. Catastrophic earthquake and tsunami events in Japan resulted in the nuclear accident that forced us to rethink our approach to nuclear safety, requirements and facilitated growing interests in designs, which can withstand natural disasters and avoid catastrophic consequences. This book is one in a series of books on nuclear power published by InTech. It consists of ten chapters on system simulations and operational aspects. Our book does not aim at a complete coverage or a broad range. Instead, the included chapters shine light at existing challenges, solutions and approaches. Authors hope to share ideas and findings so that new ideas and directions can potentially be developed focusing on operational characteristics of nuclear power plants. The consistent thread throughout all chapters is the â€œsystem-thinkingâ€ approach synthesizing provided information and ideas. The book targets everyone with interests in system simulations and nuclear power operational aspects as its potential readership groups - students, researchers and practitioners.

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