23rd ABCM International Congress of Mechanical Engineering
December 6-11, 2015, Rio de Janeiro, RJ, Brazil

Thermal-Hydraulic Analysis Under LOFA Hypothesis of a Plate-Type Fuel Surrounded by Two Water Channels Using Relap5 Code

Itamar Iliuk
EPUSP-Polytechnic School of the University of São Paulo. Av. Prof. Luciano Gualberto, travessa 3, n. 158, CEP: 05508-900 São Paulo, SP, Brazil.
itamar.iliuk@usp.br

José Manoel Balthazar
ITA-Aeronautics Technological Institute, Mechanical-Aeronautics Division. Praça Marechal Eduardo Gomes, 50 - Vila das Acácias. CEP: 12.228-900 São José dos Campos SP.
jmbaltha@ita.br

Ângelo Marcelo Tusset
UTFPR-Federal University of Technology - Paraná, Department of Engineering Science, Av. Monteiro Lobato, Km 04, s/n. CEP: 84016-210 - Ponta Grossa-PR, Brazil.
tusset@utfpr.edu.br

José Roberto Castilho Piqueira
EPUSP-Polytechnic School of the University of São Paulo. Av. Prof. Luciano Gualberto, travessa 3, n. 158, CEP: 05508-900 São Paulo, SP, Brazil.
piqueira@lac.usp.br

Abstract. *Thermal-Hydraulic analysis of plate type fuel has great importance to the establishment of safety criteria, also to the licensing of the future nuclear reactor with the objective of propelling the Brazilian nuclear submarine. In this work an analysis of a single plate-type fuel surrounding by two water channels was performed using the RELAP5 thermal-hydraulic code. To realize the simulations was proposed a plate-type fuel with the meat of uranium dioxide UO_2 sandwiched between two Zircaloy-4 plates. A PLOFA (Partial Loss of Flow Accident) was simulated to show the behavior of the model under this severe type of accident. The results show that the CHF (Critical Heat Flux) was detected in the central region along the axial direction of the plate when the right water channel was blocked.*

Keywords: *RELAP5, Thermal-Hydraulic, Critical Heat Flux, LOFA, Plate-Type Fuel.*

1. INTRODUCTION

Currently, Plate-type fuel is widely used in Research Reactors (RR) as well as in the core of nuclear submarines (SSN) (Ippolito Jr., 1990; Duarte *et al.*, 2014; Khedr, 2013; Perrota, 1999). An analysis to assess the trade-offs involved in the use of HEU (Highly Enriched Uranium) vs. LEU (Low Enriched Uranium) as SSN reactor fuel, regarding to factors as core life, core size, and reactor safety was conducted in (Ippolito Jr., 1990). It was accomplished by modeling one HEU and two LEU reactor cores for comparison. Besides, in this study it was determined the plate-type fuel with the fuel device made of uranium dioxide UO_2 sandwiched into two Zircaloy-4 plates is more qualified to support the high degree of requirements to power SSNs.

According to (Duderstadt and Hamilton, 1976), the more critical limitation is the heat flux that can be transferred from the clad to the coolant in water-cooled reactors (e.g. PWRs or BWRs). This thermal limitation, known as Critical Heat Flux (CHF), is of primary concern in water-cooled reactors cores in which the coolant temperature is allowed to approach the boiling point. The CHF has been the subject of research in the field of boiling heat transfer by nuclear engineers for many decades, (Blanchat and Hassan, 1989). The term CHF describes the value of the heat flux at which and local point at which this state of the system occurs and the Critical Heat Flux Condition (CHFC) defines the state of the system when the characteristic reduction in heat transfer coefficient has just occurred (Collier and Thome, 1996). Above certain heat flux magnitudes, the heat transfer to the coolant becomes unstable as a film of vapor forms to cover the fuel element surface. At this point the clad temperature will increase dramatically (several hundred degrees) leading to clad failure (Duderstadt and Hamilton, 1976). A study was conducted by (Blanchat and Hassan, 1989) in order to predict the behavior of the secondary side of the steam generator by using the RELAP5/MOD2 computer code and, in particular, to obtain a better prediction of CHF in bundles.

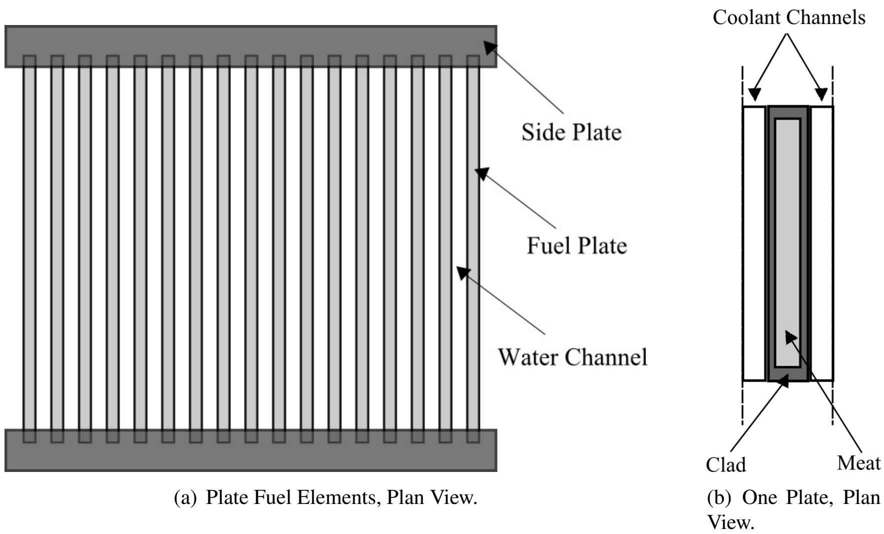
In general the transients limits in PWRs reactors to be considered include a Loss of Coolant Accident (LOCA), an overpower transient, and a Loss of Flow Accident (LOFA) (Todreas, 2006). The Loss of Flow Accident also consists of two categories, the Complete Loss of Flow Accident (CLOFA) and the Partial Loss of Flow Accident (PLOFA). LOFA simulations have been limited so far to the investigation of protected transients only (Housiadas, 2000). With the reactor shutdown system enabled, all LOFA simulations predict that clad temperature remains well below the clad melting point and that no flow instabilities take place in the cooling channels. The RR safety calculations entail usually the simulation of several selected cases within two broad accident categories, namely, re-activity insertion accidents, and LOFAs. In (Housiadas, 2000), it was investigated through numerical simulations the course of unprotected loss-of-flow transients in a typical pool-type research reactor. A sensitivity analysis of the loss of forced flow accident due to pump stop as a result of loss of the electricity supply for the IRT - Sofia was investigated by (Belousov, 2011). PARET code was used (Huda *et al.*, 2003) to analyze two most important thermal hydraulic design parameters of the 3 MW TRIGA MARK research reactor. The first design parameter is the DNBR (Departure from Nucleate Boiling Ratio), which is defined as the ratio of the critical heat flux to the heat flux achieved in the core. The second design parameter is the loss-of-flow accident scenario of the TRIGA reactor.

The RELAP5 is a highly generic code that, in addition to calculating the behavior of a reactor coolant system during a transient, can be used for simulation of a wide variety of hydraulic and thermal transients in both nuclear and non-nuclear systems, involving mixtures of steam, water, non-condensable, and solute (RELAP5/MOD3.3 Code Development Team, 2001). RELAP5 was initially designed to thermal hydraulic analysis and simulation of commercial power reactors. In recent works, the code was used to thermal hydraulic analysis of research reactors (Soares *et al.*, 2010; Reis *et al.*, 2010; Costa *et al.*, 2010; Khedr *et al.*, 2005; Koncar and Mavko, 2003).

In this work an analysis of a plate-type fuel surrounded by two water channels was performed, considering the fuel device made of uranium dioxide UO_2 and sandwiched between two Zircaloy-4 plates. The simulations were carried out using the RELAP5/MOD 3.3 to solve the equations of the theoretical model. A PLOFA was simulated to show the behavior of the model in this severe type of accident. The results show that the Critical Heat Flux Condition was reached in the system and the maximum CHF was detected in the central region of the plate.

2. THEORETICAL MODEL

In Fig. 1(a), a theoretical model of plate type nuclear fuel with 17 elements welding in 2 side plates is depicted. Also the Fig. 1(b) shows one fuel plate surrounded by two coolant channels. In this model, it was considered that the fuel device is built with uranium dioxide UO_2, sandwiched between two Zircaloy-4 plates. The heat generated into fuel was transferred through the meat cladding by conduction and through the coolant by convection (Khedr, 2013).



(a) Plate Fuel Elements, Plan View.

(b) One Plate, Plan View.

Figure 1. Theoretical Model

3. NODALIZATION MODEL

In order to analyze the thermal-hydraulic behavior of the theoretical model, a RELAP5 input deck and nodalization for the one single plate was designed. The nodalization diagram can be seen in Fig. 2. Two pipe components represent the water channels surrounding the one heat structure component. This model was divide in twelve nodes along the axial direction of the channels (Pipe component) and twelve mesh points along of the radial axis of the plate fuel (heat structure component). A trip valve component was used to simulate a flow blockage in the right channel of the system, representing a PLOFA.

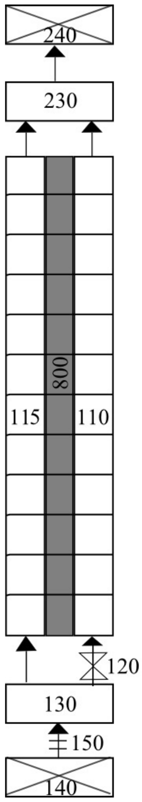


Figure 2. RELAP5 Nodalization Model.

A short description of the components with the identification numbers and types is presented in the Table 1.

Table 1. Main components of the nodalization

Component	Nodalization Element	Element type
Left Channel	115	PIPE
Right Channel	110	PIPE
Lower Plenum	130	BRANCH
Upper Plenum	230	BRANCH
Heat Structure	800	HEAT STRUCTURE
Source	140	TMDPVOL
Sink	240	TMDPVOL
Valve	120	VALVE

The main physical data for the simulations are listed in Table 2 where the measures are in millimeters, and are in accordance to (Ippolito Jr., 1990) as well as the initial conditions for the simulation. The heat generation was 30 MW and the initial flow velocity was 2.7 m/s. The system had a constant pressure of 15.0 Mpa, typical in PWR reactors. The inlet/outlet temperatures are 563 °K and 593 °K respectively. The core temperature was 583 °K. The pressure drop is equal for the two channels.

Table 2. Main physical data for the simulations

Parameter	Value	Parameter	Value
Active heat structure height(mm)	800.0	Heat structure one side surface area(mm^2)	75.45x800.0
Clad thickness (mm)	0.385	Fuel thickness (mm)	2.22
Axial power distribution	Cosine	Left channel thickness (mm)	2.63
Active heated width (mm)	75.45	Right channel thickness (mm)	2.63
Right channel width (mm)	81.04	Left channel width (mm)	81.04

4. NUMERICAL RESULTS

In this section, the results are shown considering that, in the left channel, the behavior of the plate fuel is studied without channel blockage and in the right channel, with the valve blocked to simulate a channel blockage as a PLOFA. All the simulations were carried out with time span of 1500 s. To simulate a PLOFA, the component valve was driven at 500 s of simulation time, causing the total blockage of the flow of water in the right channel.

Figure 3 presents the temperature of the heat structure, where the increase in the values occurs at 500 s of simulation. The temperature is the medium value measured between the two sides of the heat structure. The htvat-06 and htvat-07 nodes are the central points of the plate fuel, where the power generation is larger.

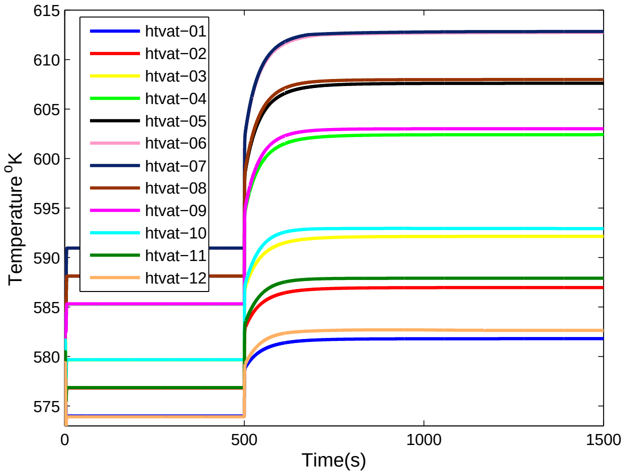


Figure 3. Temperature of the Heat Structure.

In Fig. 4 some points of the fluid temperature were picked and plotted in order to demonstrate the effect of PLOFA. The tempf-11001 is the first volume of the pipe (inlet) and tempf-11012 is the latest volume (outlet) of the right channel. At 500 s of simulation an increase of approximately around 20 °K occurs. On the other hand, the left channel temperature do not presents significant variation. In the pipe 115, the inlet and outlet values remained near to the initial condition temperature (tempf-14001).

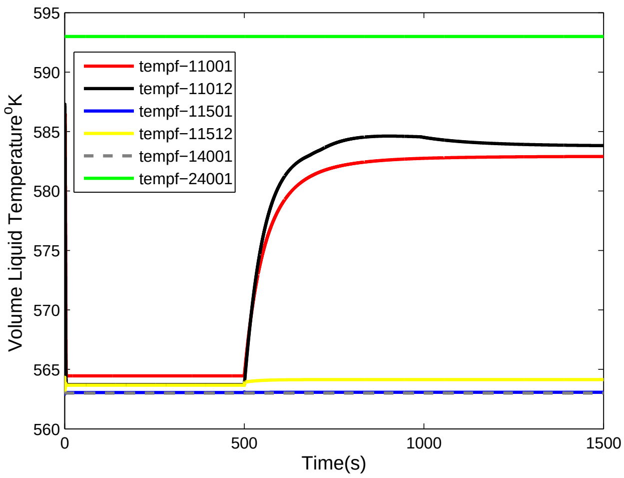


Figure 4. Volume Liquid Temperature.

The heat transfer coefficients in the Fig. 5, present a great variation in the case of a right channel blockage at 500 s of simulation. The hthtc-800100100 and hthtc-800101200 shows the heat transfer coefficients values of the heat structure over the left surface area, where its behavior remains stable. The hthtc-800100101 and hthtc-800101201 shows the heat transfer coefficients values of the heat structure over the right surface area, where an abrupt decrease in the heat transfer coefficients occurs in the case of PLOFA.

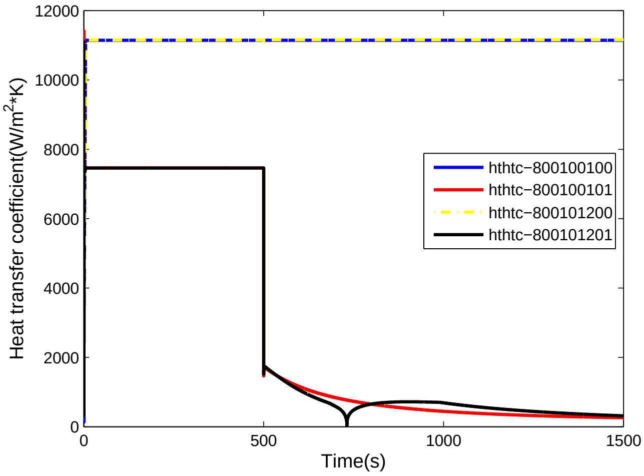


Figure 5. Heat Transfer Coefficients.

The Critical Heat Flux occurred in the simulation is depicted in the Fig. 6, where the htchf-800100600 and htchf-800100700 are the central mesh points in the left side of the heat structure, and no CHF occurs along the time of simulation with the normal flow in the channel. On the other hand, where the htchf-800100601 and htchf-800100701 are the central mesh points in the right side of the heat structure, the channel blockage causes the heat structure temperature to increase, at the point where the CHF occurs. In this condition of instability, the fuel heat could not be retired and the melting point of the fuel could be reached.

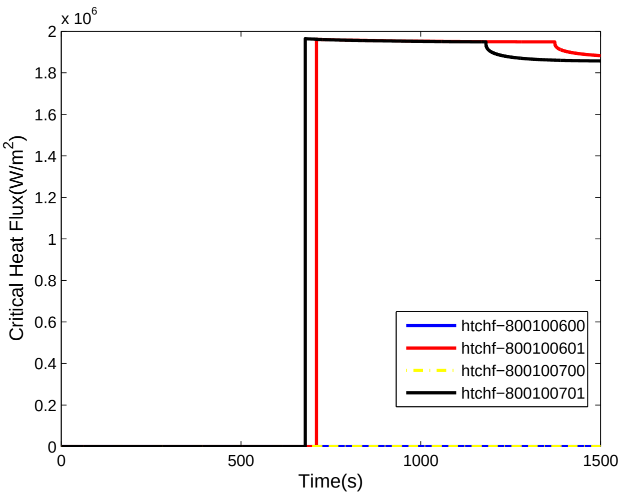


Figure 6. Critical Heat Flux.

The Critical Heat Flux Ratio (CHFR) in Fig. 7 is the ratio between the Critical Heat Flux and the local Heat Flux. This term is a measure of the margin to Departure from Nucleate Boiling (DNB), an important criterion to limitation the maximum power of the reactor core design. In this case, the htchfr-800100600 and htchfr-800100700 (left side surface area), presents the value of 0 and htchfr-800100601 and htchfr-800100701 (right side surface area) achieved larger values after the channel blockage at 500s of simulation. It could be occurred due to the application of a chopped cosine profile of power to the heat structure.

In the Fig. 8 is depicted the distribution of fuel plate temperature when the flow of the coolant is normal for both channels. Along the fuel thickness axis, the distribution of the temperature over the twelve mesh points can be verified. The intense red color represents the central region of the plate. The cosine profile of imposed power model the curve data distribution. At the height of the active heat structure, the distribution of temperature likewise in the thickness remains

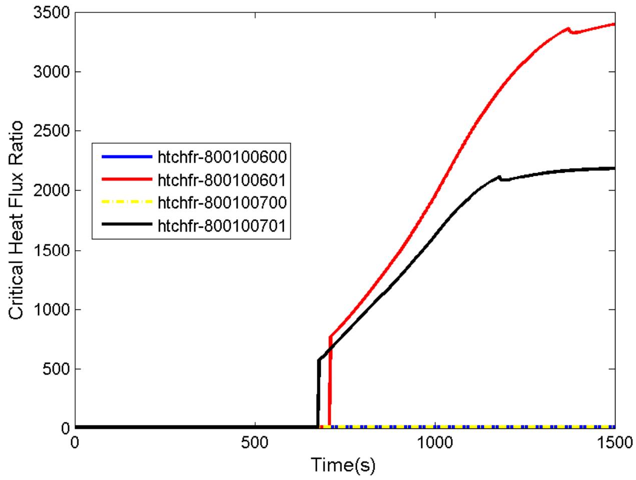


Figure 7. Critical Heat Flux Ratio.

concentrated in the central volumes of the both channels. The Matlab datatips depicted on the borders, shows the temperature in the surface area on the left and on the right side of the fuel plate. The temperature presents an equal value of 582.9 °K. It is achieved because the coolant flows normally along the channels.

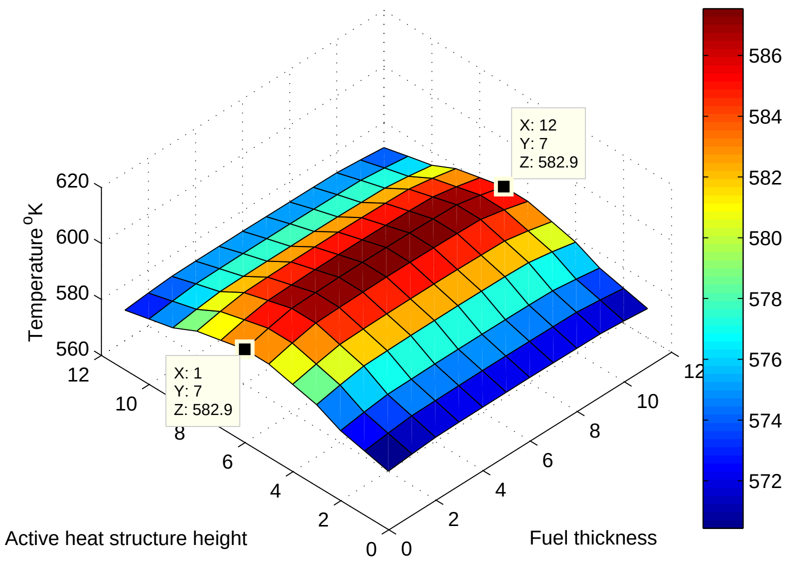


Figure 8. Fuel Plate Temperature Distribution.

In the Fig. 9 is depicted the fuel plate temperature distribution when the flow of the coolant is blocked in the right channel to simulate the hypothesis of a PLOFA scenario. Along the fuel thickness axis, the distribution of the temperature over the twelve mesh points, presents a growth in the direction of the right surface area of the plate fuel. The intense red color that represents the larger temperature is depicted in the extreme mesh point at the right region of the plate with the value of 615.8 °K. The profile of power distribution no longer presents a cosine form. In the height of the active heat structure, the temperature distribution no longer presents equally concentration in the central volumes of the both channels. The temperature presents a value of 602.5 °K in the left surface of the heat structure. Clearly, the blockage of the right channel coolant flow also causes the elevation of the temperature in the left channel because of the conduction of the heat from one side to the other of the plate.The increase of the temperature of the heat structure for the case where the right channel was blocked achieves 5.62% higher rather than the case without the channel blockage.

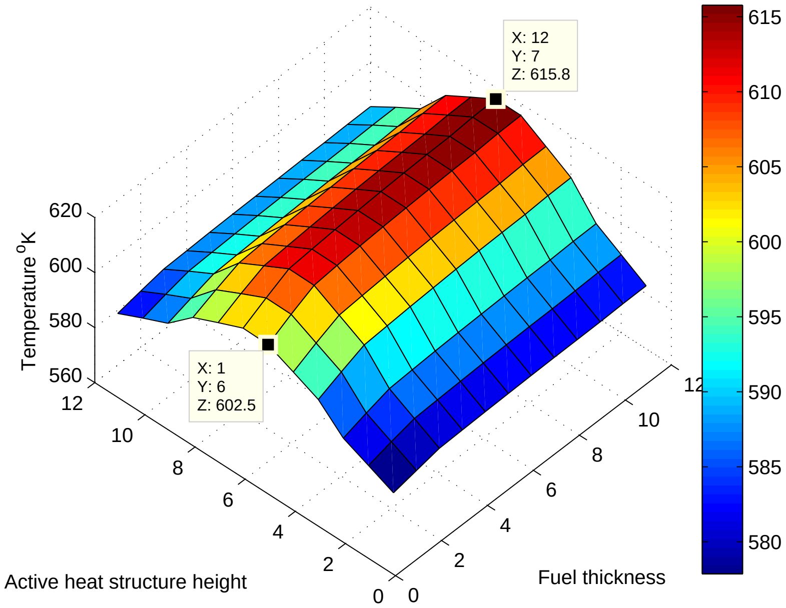


Figure 9. Fuel Plate Temperature Distribution.

5. CONCLUSIONS

Thermal-Hydraulic analysis is of fundamental importance to conduct nuclear experiments and the RELAP5 code is a powerful tool to perform such analysis. In this work the analysis of a single plate-type fuel surrounded by two water channels shows the capabilities of the RELAP5 code to perform transient analysis and to simulate a PLOFA severe accident. As shown, when the valve blocked the water flow the heat removal by the coolant was decreased and the CHF has occurred on the right surface of the heat structure. The increase of temperature on the right side has influenced the temperature of the left side by heat conduction, but no hot spots were detected on the left side of the plate. In the real scenario this could be the cause of the melting of the core.

6. ACKNOWLEDGEMENTS

The authors acknowledge support by FAPESP, CNPq and CAPES, all Brazilian research funding agencies.

7. REFERENCES

Belousov, S.I., 2011. "Safety analyses of irt - sofia leu core: Sensitivity analyses of loss of flow accident". In *RERTR 2011-33rd International Meeting on Reduced Enrichment for Research and Test Reactors*. Santiago, Chile.

Blanchat, T. and Hassan, Y., 1989. "Comparisons of critical heat flux correlations with bundle flows". In *Annual Meeting of the American Nuclear Society*. Atlanta, GA.

Collier, J.G. and Thome, J., 1996. *Convective Boiling and Condensation*. Oxford engineering science series. Clarendon Press.

Costa, A., Reis, P., Pereira, C., Veloso, M., Mesquita, A.Z. and Soares, H., 2010. "Thermal hydraulic analysis of the ipr-r1 triga research reactor using a relap5 model". *Nuclear Engineering and Design*, Vol. 240, No. 6, pp. 1487–1494.

Duarte, J., Iliuk, I. and Piqueira, J., 2014. "Thermal-hydraulic analysis of plate fuel small pressurized nuclear reactors under a loss of flow accident". In *Proceedings of the VIII Congresso Nacional de Engenharia Mecânica - CONEM*.

Duderstadt, J. and Hamilton, L., 1976. *Nuclear Reactor Analysis*. John Wiley and Sons, Inc.

Housiadas, C., 2000. "Simulation of loss-of-flow transients in research reactors". *Annals of Nuclear Energy*, Vol. 27, No. 18, pp. 1683 – 1693.

Huda, M.Q., Mahmood, M., Chakrobortty, T., Rahman, M. and Sarker, M., 2003. "Analysis of the dnb ratio and the loss-of-flow accident (lofa) of the 3 mw triga mark ii research reactor". Technical report, Institute of Nuclear Science and Technology Atomic Energy Research Establishment.

Ippolito Jr., T.D., 1990. *Effects of Variation of Uranium Enrichment on Nuclear Submarine Reactor Design*. Master's thesis, Massachusetts Institute of Technology, Cambridge, MA.

Khedr, A., 2013. "Thermal-hydraulic analysis of heated plate under asymmetrical cooling conditions". *Arab Journal of Nuclear Science and Applications*, Vol. 46, No. 3, pp. 219–230.

Khedr, A., Adorni, M. and D'Auria, F., 2005. "The effect of code user and boundary conditions on relap calculations of mtr research reactor transient scenarios". *Nuclear Technology and Radiation Protection*, Vol. XX, No. 1, pp. 16–22.

Koncar, B. and Mavko, B., 2003. "Modelling of low-pressure subcooled flow boiling using the {RELAP5} code". *Nuclear Engineering and Design*, Vol. 220, No. 3, pp. 255 – 273.

Perrota, J., 1999. *Proposta de um Núcleo de Reator PWR Avançado com Características Adequadas para o Conceito de Segurança Passiva*. Ph.D. thesis, IPEN - Universidade de São Paulo, São Paulo.

Reis, P., Costa, A., Pereira, C., Veloso, M., Mesquita, A.Z., Soares, H. and Barros, G.P., 2010. "Assessment of a relap5 model for the ipr-r1 triga research reactor". *Annals of Nuclear Energy*, Vol. 37, No. 10, pp. 1341–1350.

RELAP5/MOD3.3 Code Development Team, 2001. *RELAP5/MOD3.3 Code Manual*. Idaho National Laboratory, Idaho Falls, ID (USA).

Soares, H., Aronne, I., de Almeida, Y., Lima, C., Costa, A., Veloso, M. and Reis, P., 2010. "Thermal hydraulic analysis of the multipurpose research reactor osiris using a relap5 model". In ABCM, ed., *Proceedings of ENCIT 2010*. Uberlandia, MG.

Todreas, N., 2006. "Thermal hydraulics design limits - class note ii". Technical report, Massachusetts Institute of Technology.

8. RESPONSIBILITY NOTICE

The authors are the only responsible for the printed material included in this paper.