THERMAL-HYDRAULIC ANALYSIS OF THE IEA-R1 RESEARCH REACTOR – A COMPARISON BETWEEN IDEAL AND ACTUAL CONDITIONS

Pedro Ernesto Umbehaun
IPEN - Instituto de Pesquisas Energéticas e Nucleares
umbehaun@ipen.br

Walmir Maximo Torres
IPEN - Instituto de Pesquisas Energéticas e Nucleares
wmtorres@ipen.br

Abstract. *Thermal-hydraulic analysis were performed for the IEA-R1 research reactor considering ideal, estimated and actual flow rate conditions through the fuel elements. The ideal conditions were obtained dividing the total primary flow rate among the fuel elements and the estimated conditions were calculated using the computer program FLOW. The actual flow rate conditions were experimentally measured using an instrumented dummy fuel element. The results show that the actual conditions are far from ideal and calculated ones due to the high bypass flow that deviates the active reactor core through the irradiation devices, gaps, couplings, etc..Thus, the safety margins are smaller for the actual flow conditions.*

Keywords*. research reactor, thermal-hydraulic, safety*

1. Introduction

The IPEN IEA-R1 is a 5 MW pool type research reactor that uses MTR (Material Testing Reactors) fuel elements in the core. Each fuel element has 18 fuel plates assembled on two lateral support plates, forming 17 independent closed flow channels. The safe operation of the reactor is guaranteed maintaining suitable safety margins in any operational conditions. These safety margins (DNBR, ONB, CHF and maximum surface temperature) are verified in the thermal-hydraulic analysis (THA´s) of the core. To perform the THA it is necessary to know some parameters, such as: heat flux distribution, geometric characteristics, material properties and flow rates through the fuel elements. The uncertainties of these parameters are also necessary.

The flow rate through the fuel elements is an important parameter and it is difficult to determine due to the geometric complexity of the core. The International Atomic Energy Agency in IAEA - TECDOC 233 (1980) suggests that the flow rate through the fuel elements is the total reactor primary flow rate divided by the number of fuel elements (ideal condition). This value is far from the actual one because the core has fuel elements and other components such as: reflectors, irradiators, plugs and still secondary bypass holes, gaps and couplings. A more realistic value of the flow rate is obtained using the computer program FLOW that uses experimental and theoretical correlations for components pressure drop, holes and other flow paths in the calculation. However, this calculated value can be still far from actual one. A dummy fuel element (DMPV-01) was designed and constructed to measure the flow rate distribution at the core of the IEA-R1. It is made of aluminum in natural size and has static pressure taps at the inlet and the outlet region, and a dynamic pressure tap at the outlet nozzle. The measured values show that the actual flow through the fuel elements is lower than the values obtained by two methods and, therefore, the bypass flow is greater than the expected. After investigations, some causes of undesired bypasses were identified and actions were taken to reduce or eliminate them.

This work presents a comparison among the results of the three performed THA´s for 5 MW reactor operation power. In the first THA the ideal conditions of the flow rate distribution on the core were considered, in the second, the calculated flow rates obtained from FLOW computer program were used, and in the third, the measured flow rates made by a dummy fuel element were used.

2. IEA-R1 research reactor

The IEA-R1 is a pool type, light water cooled and moderated research reactor that uses MTR fuel elements, Fig. 1. The reactor is located at IPEN (Instituto de Pesquisas Energéticas e Nucleares) in São Paulo and was designed and built by Babcock and Wilcox Co. in 1957. IEA-R1 research reactor is a multipurpose research reactor. It has been used for basic and applied researches, training and mainly for radioisotope production for applications in medicine. In 1995, in view of a favorable budget from Federal Government and the priorities given to the production of some useful radioisotopes, IPEN took the decision to modernize and upgrade the reactor power operation from 2 to 5 MW and increase its operational cycle from 8 hours/day, 5 days a week to 120 hours/week in continuous operation. In order to optimize the neutron flux and to have enough reactivity for continuous operation, the size of the active core was changed from 30 to 24 fuel elements, Fig. 2. The reactor primary system has a pool with the core containing the fuel elements and other components, and has two pumping circuits to promote the down flow through the core. The reactor primary system also has: a decay tank to reduce the N_{16} activity, two heat exchangers to remove the generated heat in the core, a flow rate measurement system (flow nozzle and differential pressure transmitter) and a inlet flow distributor at the pool to prevent preferential flow paths and waves.

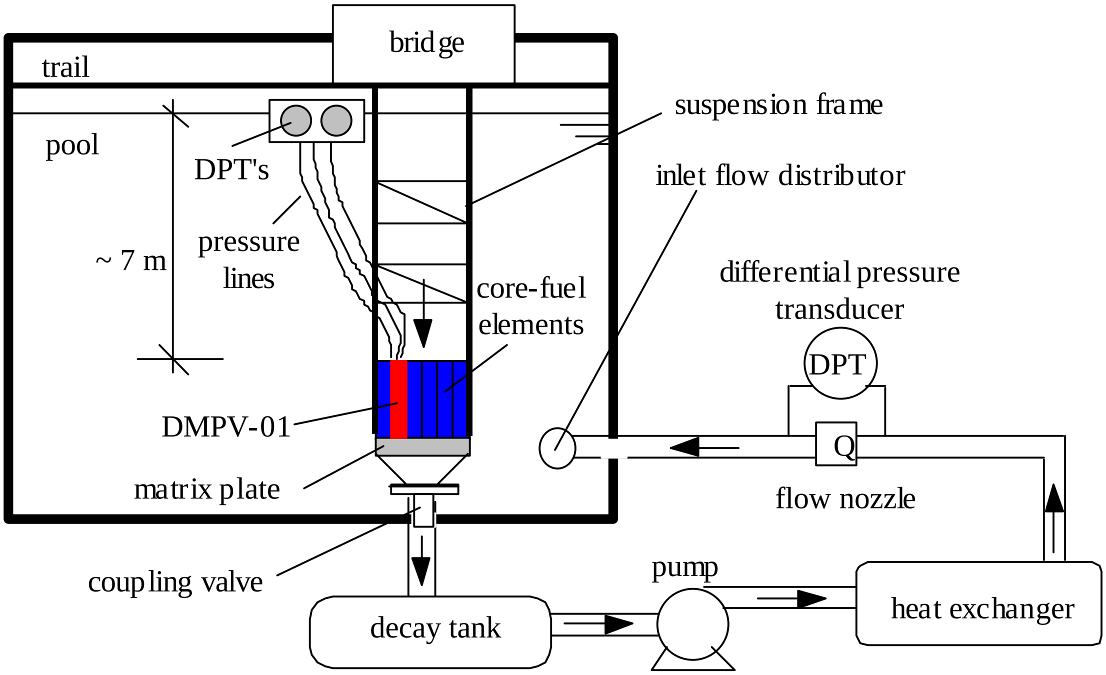

Figure 1. Simplified draw of the IEA-R1reactor primary system.

ΔP	DP	DP	DP	DP	DP	DP	SP
SP	SP	SP	SP	SP	SP	NS	R
R	SP	R	EIRA	R	R	R	R
EIS	EIS	R	EIRA	R	GI	R	R
EIS	EIS	FE 153	FE 168	FE 156	FE 160	FE 150	R
R	EIGRA I	FE 158	CFE 166	FE 169	CFE 180	FE 171	EIF
R	R	FE 164	FE 161	EIBE	FE 162	FE 163	R
R	EIGRA II	FE 159	CFE 179	FE 170	CFE 167	FE 154	R
R	R	FE 152	FE 155	FE 157	FE 165	FE 151	R
R	R	R	R	R	R	R	R

LEGEND

ΔP = core pressure drop measurement
DP = double plug
SP = single plug
NS = neutron source
R = graphite reflector
FE = fuel element
CFE = control fuel element
EIGRA´s = irradiators
EIS = irradiator
EIBE = irradiator
GI = irradiator
EIF = irradiator
EIRA = irradiator

Figure 2. IEA-R1 core components and DMPV-01 flow rate measurement positions FE´s (153, 169, 170 e 152).

3. Neutronic model

The heat fluxes or power density in the reactor core were calculated using CITATION (Fowler et all (1972)), MCNP–4C (Briemeister, 2000), HAMMER (Barhen, 1978) and LEOPARD (Barry, 1963 and Kerr et all (1991)) neutronic codes, and depends on the power operation, burn-up, number of fuel elements, etc.. Heat fluxes are not uniform and depend on the axial and radial positions in the reactor core. The THA´s are performed to verify the core safety limits, thus the worst conditions (hot channel conditions) are used in the calculations. Figure 3 shows the axial hot channel conditions resulting of the calculations for 5 MW reactor operation power. The calculated axial power peaking factor is 2.73, corresponding to a local heat flux of q = 63.53 W/cm^2.

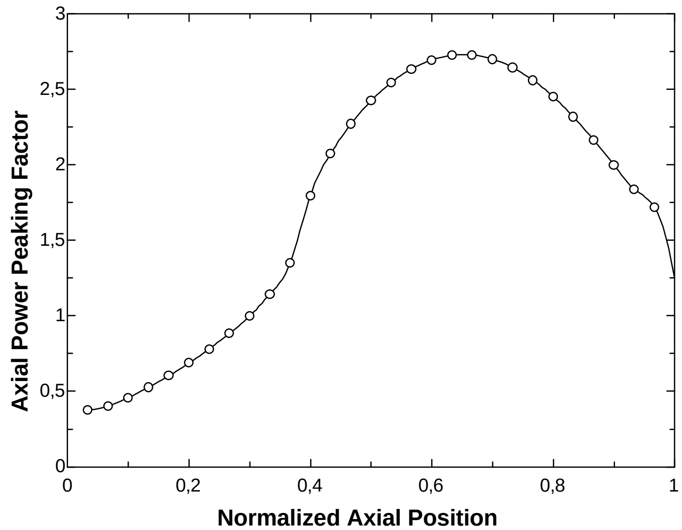

Figure 3. Axial heat flux profile of the core hot channel.

4. Thermal hydraulic model

The THA´s are performed using the conduction and convection heat transfer equations for the rectangular channels formed by fuel plates with coolant fluid down flow, as shown in Fig. 4. The following heat changes were considered in the model: a) transversal heat conduction in the fuel plates, b) convection from cladding surface to the fluid fow and c) enthalpic transport due fluid flow. Axial conduction in the fuel plates and coolant fluid were not considered in the calculations. Additional information about the methodology used in the THA´s can be obtained in Umbehaun (2000). The system formed by the coupled equations was solved, for steady state condition, using EES (Engineering Equation Solver) software developed by Klein et all (2000). The geometry and involved materials must be known and their uncertainties. The flow rates in the internal channels were considered constant and uniformly distributed among them.

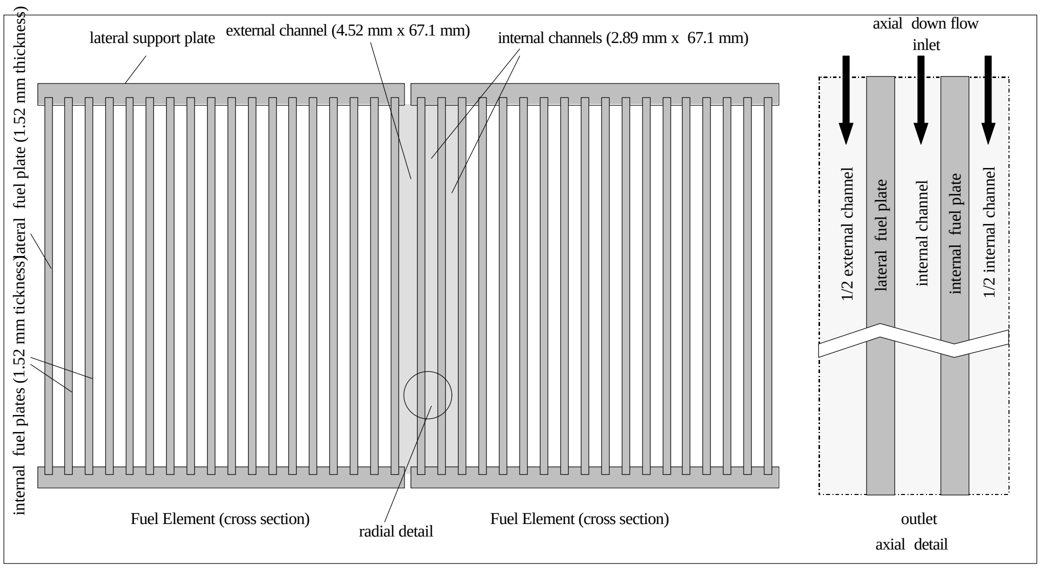

Figure 4. Thermal model sketch – two fuel elements.

4.1.Thermal-hydraulic safety margins – definitions and correlations used

ONB Temperature – Onset of Nucleate Boiling Temperature

Onset of Nucleate Boiling is taken as a limit for single-phase cooling and is not a limiting criterion in design of a fuel element. It is the heat transfer regime that should be clearly identified for proper hydraulic and heat transfer considerations, i. e., single-phase flow versus two-phase flow. The nucleate boiling occurs at a wall temperature over T_{sat} by a quantity $T_w - T_{sat}$. Under **ONB** conditions, the fuel wall surface temperature (**ONB Temperature**) over which nucleate boiling will occur for a given local coolant pressure and surface heat flux can be expressed by the correlation given by the Eq. (1) developed by Bergles et all (1964).

$$T_w = T_{sat} + \frac{5}{9}\left(\frac{9.23q}{P^{1.156}}\right)^{\frac{P^{0.0234}}{2.16}} \tag{1}$$

where T_w is the fuel wall temperature (^oC) at ONB conditions, T_{sat} is the local saturation temperature, P is the local pressure (bar) and q is the local heat flux (W/cm^2).

DNB – Departure from Nucleate Boling;
CHF – Critical Heat Flux;
DNBR – Departure from Nucleate Boling Ratio; and
MDNBR – Minimum Departure from Nucleate Boling Ratio

Departure from Nucleate Boiling (**DNB**) is the phenomena that occur on boiling process where the density of bubbles on the heated surface becomes so large that they coalesce and form a vapor film, insulating the heated surface. Heat transfer must then take place by a combination of conduction and radiation across the vapor film. Neither of these two processes involved in film boiling is very effective, hence the heat flux decrease considerably, even when the temperature difference is increasing. In this condition the surface temperature can become too high and melting can occurs. The heat flux immediately before the occurrence of **DNB** is called critical heat flux (**CHF)**.
Departure from Nucleate Boiling Ratio (**DNBR**) is the ratio between the critical heat flux q_c (**CHF**) and the local heat flux q. Labuntsov (1960) and Mirshak et all (1959) developed correlations to predict the critical heat flux.

Labuntsov correlation for critical heat flux is given by Eq. (2) to (4).

$$q_c = 145.4q(p)\left[1 + 2.5V^2/q(p)\right]^{1/4}\left(1 + 15.1C_p\Delta T_{sub}/lP^{1/2}\right), \tag{2}$$

$$q(p) = 0.99531P^{1/3}\ (1 - P/P_c)^{4/3} \text{,and} \tag{3}$$

$$\Delta T_{sub} = T_{sat} - T_{in} - \Delta T_c \tag{4}$$

where: q_c is the critical heat flux (W/cm^2); V is the flow velocity (m/s); P is the pressure at the channel outlet; P_c is the critical pressure; λ is the heat of vaporization; ΔT_{sub} is the sub-cooling,; T_{sat} is the saturation temperature; T_{in} is the fluid inlet temperature; and ΔT_c is the temperature increase in the channel. These equations are valid in the following range of parameters: velocity (0.7 - 45 m/s); absolute pressure (1 – 200 bar); temperature sub-cooling (0 – 240^{o}C); and critical heat flux (116 – 5234 W/cm^2).

Mirshak correlation for critical heat flux is given by Eq. (5).

$$q_c = 151(1 + 0.1198\ V)(1 + 0.00914\Delta T_{sub})(1 + 0.19P) \tag{5}$$

where: v is the flow velocity (m/s); P is the absolute pressure at the channel outlet (bar); and ΔT_{sub} is the subcooling. Eq. (5) is valid in the following range of parameters: velocity (1.5 – 13.7 m/s); absolute pressure (1.72 – 5.86 bar); temperature subcooling (5 – 75^{o}C); equivalent diameter (5.3 – 11.7 mm); and critical heat flux (284 – 1022 W/cm^2).

Minimum Departure from Nucleate Boiling (**MDNBR**) is a core design criteria adopted to guarantee the safety margin for prevent critical heat flux occurrence and is related with **DNBR**, i. e., **MDNBR** is the minimum value of **DNBR** accepted in the core design. In PWR power reactors the **MDNBR** criteria is about 1.3 while for IEA-R1 research reactor was adopted **MDNBR** = 2.0. This meaning that the core reactor must operate in conditions up to those have guaranteed a minimum safety margin of 100% for critical heat flux occurrence.

FI – Flow Instability
FIR – Flow Instability Ratio

Flow instability (**FI**) refers to flow oscillations of constant or variable amplitude that are analogous to vibrations in mechanical systems. In this connection the relationship among heat flux, flow rate and pressure drop plays an important role. Flow oscillations are undesired for several reasons (mechanical vibrations of components, problems with system control, affect the local heat transfer characteristics, etc..). Flow instabilities for different geometries has been studied by several researchers. Whittle et all (1967) suggest the Eq. (6) and (7) for calculation of the mean heat flux to onset of flow instability q_{FI}.

$$\overline{q_{FI}} = r \, \frac{R c_p V D_h}{4 L_c} \left(Tsat - Tin \right) \tag{6}$$

$$R = \frac{1}{1 + h\left(\frac{D_h}{Lc} \right)} \tag{7}$$

where: q_{FI} is the mean heat flux to onset of flow instability (W/cm^2); ρ is the fluid density (g/cm^3); c_p is the specific heat (J/cm ^{o}C); V is the flow velocity (cm/s); D_h is the equivalent hydraulic diameter (cm); L_c is the heated length of channel (cm); T_{sat} is the saturation temperature at the channel outlet (^{o}C); T_{in} is the channel inlet fluid temperature channel (^{o}C); and η is a coefficient.

The ratio between the mean heat flux for onset of flow instability q_{FI} and channel mean heat flux is called flow instability ratio (**FIR**) and represents how far the operation conditions are from instability flow occurrence region. It was adopted **FIR** = 2.0 for the IEA-R1.

4.2. Flow rate - Ideal conditions (TEC-DOC 233)

The IAEA TEC-DOC 233 suggests that the flow rate through the fuel element is equal to the total primary flow rate divided by the number of fuel elements (ideal condition). The total primary flow rate of the IEA-R1 research reactor is 3000gpm (0.19 m^3/s) and the reactor core has 24 fuel elements. In this case, the flow rate through each fuel element is 7.92 x 10^{-3} m^3/s (28.4 m^3/h).

4.3. Flow rate - Calculated using FLOW

The FLOW computer program was developed to calculate the flow rate through the components of the core and flow paths. It is based on experimental and theoretical pressure drop correlations for these components and flow paths, and thus depends on the core configuration, i.e., number of fuel elements, irradiators with or without samples, secondary holes opened or closed in the matrix plate, channels formed between two fuel elements, etc.. The equations used in FLOW assume that all the components form parallels closed channels and presents the same core pressure drop. The sum of these individual flow rates is equal to the total primary flow rate. Table 1 shows the calculation results for some core configurations. The second row presents the results of today core configuration and the others were carried out for parametric study. Comparing the calculated fuel element flow rate 5.51 x 10^{-3} m^3/s (19.8 m^3/h) with the value obtained in previous section 7.92 x 10^{-3} m^3/s, we can observe a difference of approximately 44%.

4.4. Flow rate - Measured with DMPV-01

An instrumented dummy fuel element DMPV-01, Fig. 5, was designed and constructed to measure the flow rate through the fuel elements of the IEA-R1 reactor core. This element is made of aluminum in actual size and has taps to measure static and dynamic pressures. It was calibrated in an experimental circuit, Fig. 6, and a pressure drop versus mass flow rate curve, was obtained, Fig. 7. The flow rate in the experimental circuit was measured by a system with an orifice plate and a differential pressure transducer. Dummy pressure drops were measured by two differential pressure transducers. All the pressure transducers were calibrated and a type K thermocouple was used to temperature measurements for fluid properties corrections. After calibration, DMPV-01 was used to measure the flow rate in four representative fuel elements positions, as shown in Fig. 2. Figure 8 presents the measurement results and shows that the core flow rate distribution is quite uniform, Torres et all (2001). However, the measured value of 4.22 x 10^{-3} m^3/s (15.2 m^3/h) was smaller than the calculated by FLOW, indicating a bypass flow higher than the expected. After investigations using an inspection system with an underwater video camera, two problems were detected: a) an excessive flow rate through irradiator EIS; and b) some core components were not well fitted in the matrix plate. Corrective actions were taken and new measurements were performed. The results also are shown in Fig. 8 and are near to the calculated by FLOW program.

Table 1. Core flow rate distribution for 24 fuel elements, 4 control fuel elements and 0.19 m^3/s total primary flow rate.

(*) SHMP	EIRA	EIS	EIBE	core pressure drop [Pa]	FE $[m^3/s]$ x 10^{-3}	CFE $[m^3/s]$ x 10^{-3}	EIRA $[m^3/s]$ x 10^{-3}	EIB $[m^3/s]$ x 10^{-3}	EIS $[m^3/s]$ x 10^{-3}	(*) SHMP $[m^3/s]$ x 10^{-3}	(*) CBFE $[m^3/s]$ x 10^{-3}	(*) CBRI $[m^3/s]$ x 10^{-3}
13	3	1	1	8540	5.20	4.79	3.30	5.39	2.62	1.53	0.725	0.172
6	3	1	1	9530	5.51	5.08	3.49	5.70	2.77	1.62	0.769	0.177
6	3	1	0	10100	5.68	5.25	3.59	-	2.85	1.66	0.792	0.181
0	3	1	1	10520	5.81	5.37	3.66	6.00	2.91	1.66	0.811	0.183
0	3	1	0	11180	6.00	5.54	3.85	-	3.00	-	0.839	0.187
0	3	0	0	11520	6.10	5.64	3.83	-	-	-	0.853	0.189
0	2	0	0	11970	6.22	5.76	3.91	-	-	-	0.869	0.193
0	1	0	0	12440	6.36	5.88	3.99	-	-	-	0.889	0.196
0	0	0	0	12950	6.49	5.93	-	-	-	-	0.908	0.199
0	0	1	0	12540	6.38	5.91	-	-	3.18	-	0.892	0.196

(*) SHMP – opened secondary holes in the matrix plate; CBFE – channels between two fuel elements; CBRI – channels between reflector and irradiator

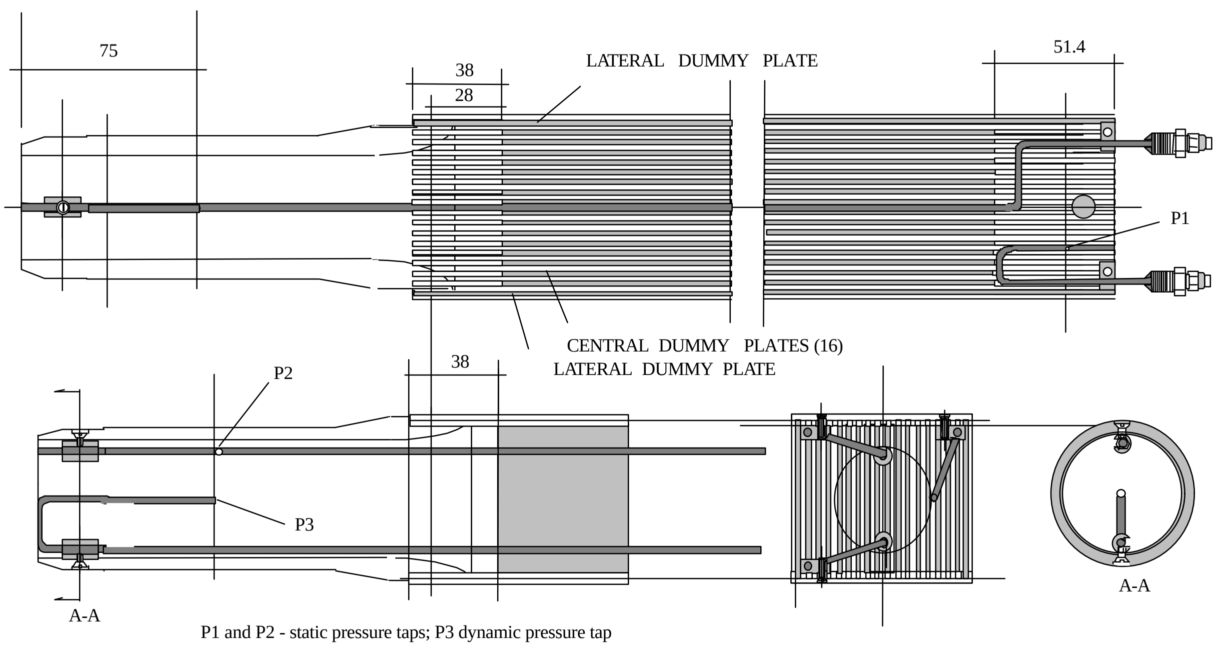

Figure 5. Instrumented dummy fuel element

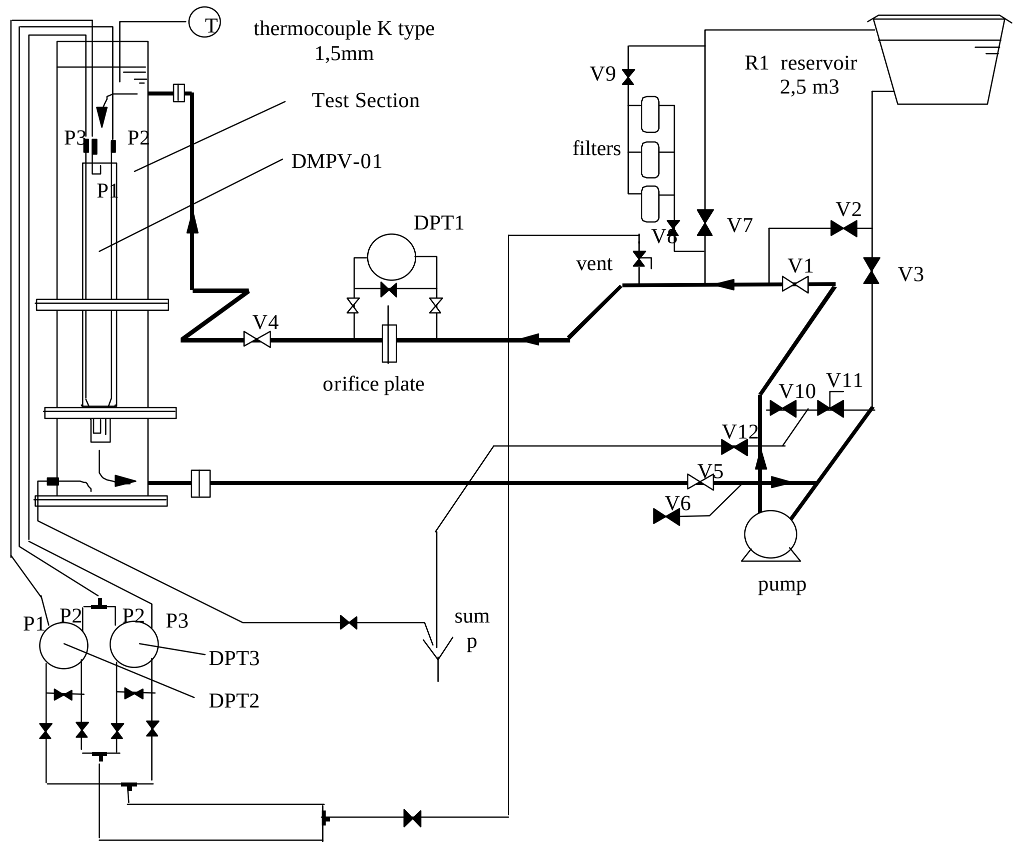

Figure 6. Experimental circuit used for DMPV-01 calibration

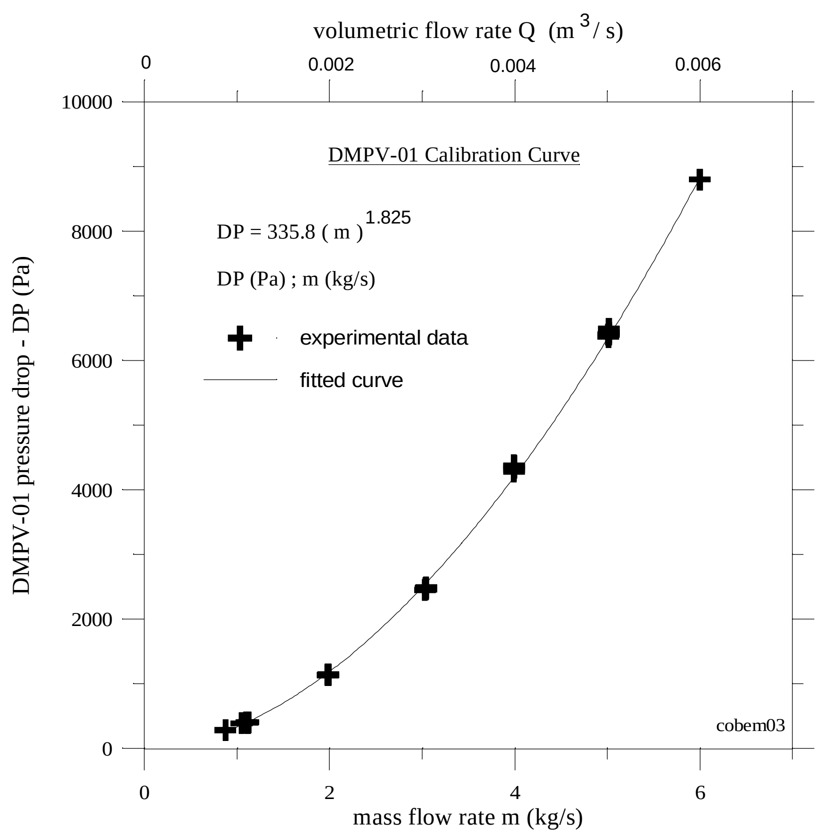

Figure 7. DMPV-01 calibration curve.

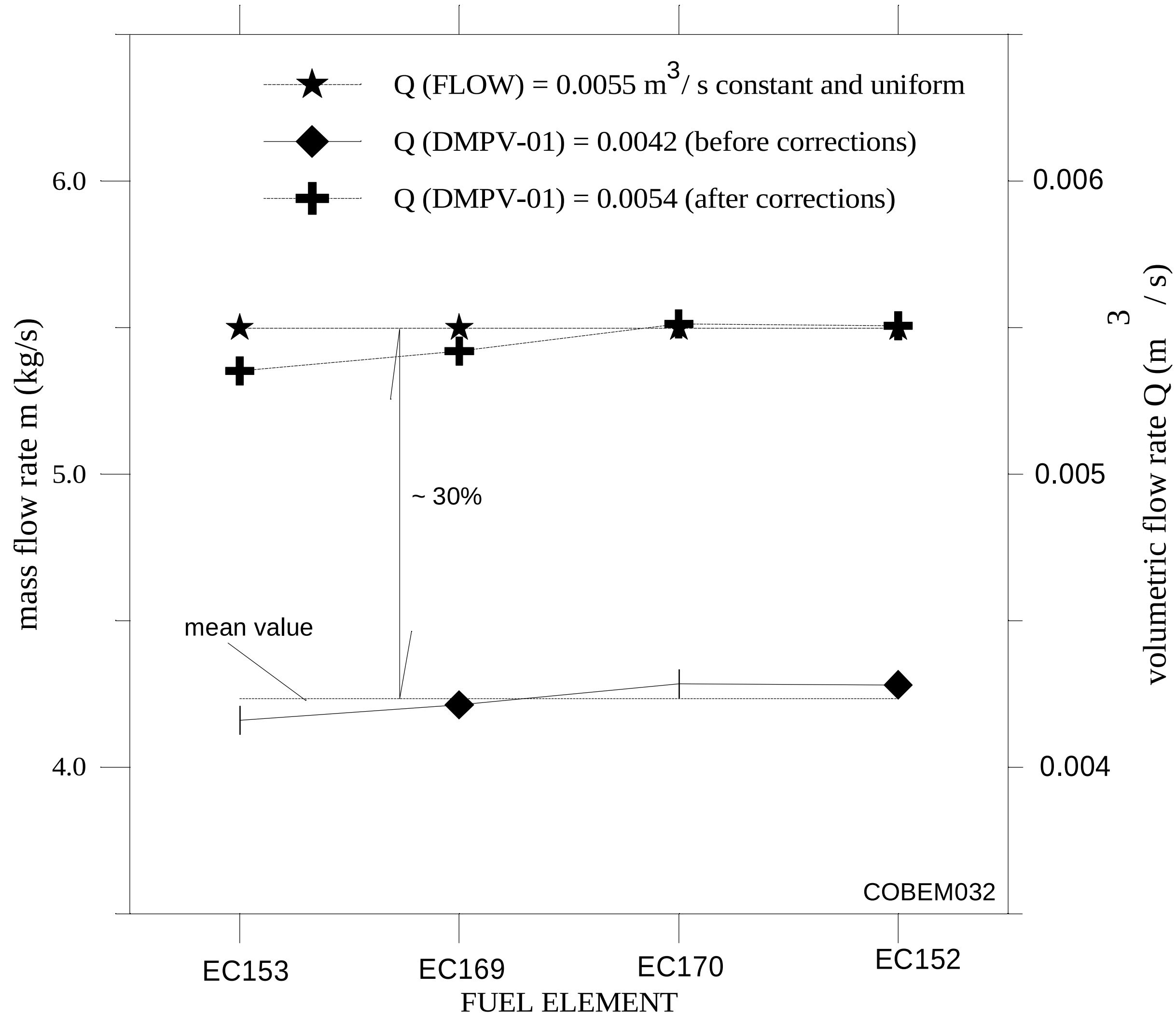

Figure 8. Flow rate measurements with DMPV-01.

5. Results

Figures (9) to (11) show the results of the THA´s in a parametric study performed to verify the influence of the flow rate in the thermal-hydraulic safety margins of the IEA-R1 research reactor for 5 MW operation power. Figure (9) shows the surface temperature along the axial position in the hot channel and the correspondent local ONB temperature. One can see that the surface temperature is near to ONB temperature for the measured flow rate 15.5 m^3/h (without corrective actions), while the surface temperature for ideal flow conditions is far from the ONB temperature. The results using the flow rate calculated by the FLOW computer program and measured values after corrective actions produced intermediate values. One can conclude that to use ideal flow conditions suggested by IAEA – TECDOC 233 in the calculations induces in erroneous way to higher safety margins.

Figure (10) shows the behavior of FIR and DNBR for flow rates between 15.5 m^3/h and 28.5 m^3/h. One can see that the lower values for FIR (~4.5) and DNBR (~4.5) were obtained for the minimum flow rate 15.5 m^3/h (without corrective actions). For ideal flow conditions FIR (~ 8.5) and DNBR (~ 6.5); and for calculated flow rate by FLOW and corrected measured flow rate by DMPV-01 FIR (~5.5) and DNBR (~5.0).. Figure (11) shows the peak surface temperature, ONB temperature and exit flow temperature for flow rates from 15.5 to 28.5 m^3/h. Here, one can observe again the reduction in the safety margins between ideal and actual flow conditions. It is good remind that these safety margins are still high for the minimum flow rate.

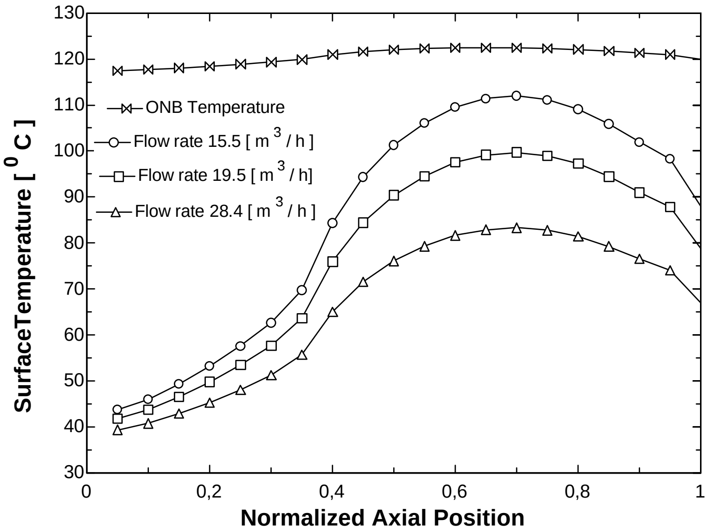

Figure 9. Surface temperature along the hot channel for different flow rate conditions.

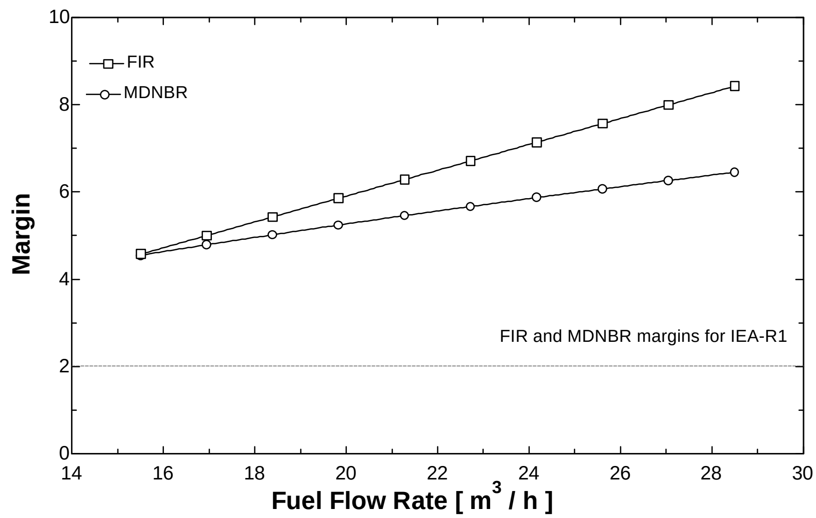

Figure 10. FIR and MDNBR safety margins versus Fuel flow rate.

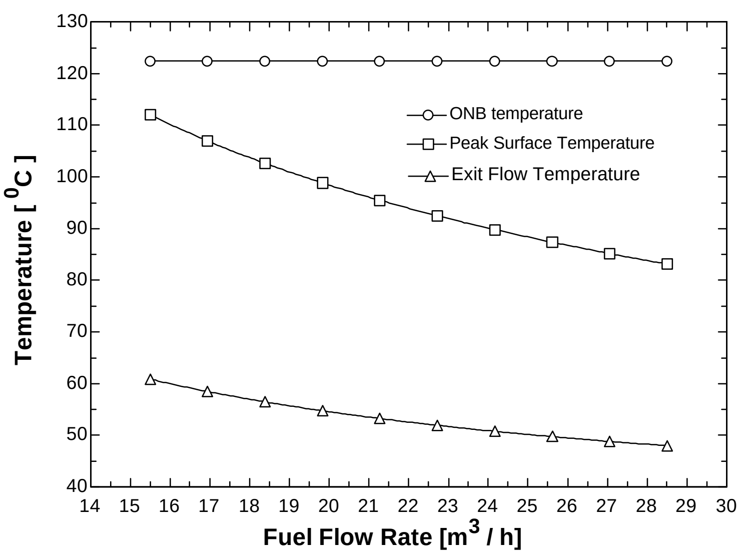

Figure 11. Peak surface temperature, exit flow temperature and ONB temperature versus fuel flow rate.

6. Conclusions

The THA´s results show that all the safety margins calculated using the actual flow rate conditions are lower than those calculated with the ideal conditions. It is good remind that these safety margins are still high for the minimum flow rate. Based on this one can conclude: a) The calculation suggested by IAEA TECDOC-233 is not a good approximation for high reactor power operation and for cores with complex geometries; b) It is very important to reactor operation have a tool, in our case the DMPV-01dummy element, to permit flow rate measurements through the fuel elements because the core bypass flow can be higher than the expected and the pressure drop correlations can not be suitable to use in this specific case. For example, the first measured values of flow rates indicated an excessive bypass flow rate from active core and some corrective actions were taken to correct them; c) Sub-aquatic inspection systems also are important to perform core inspections. These inspections showed some graphite reflectors and irradiators not well fitted on the matrix plate and an irradiator had to be modified to reduce the bypass flow; and d) All new irradiators must be designed and experimentally tested to evaluate the core flow impact before to be assembled in reactor core.

7. References

Barhen, J., Rhotenstein, W., Taviv, E., 1978, “The HAMMER Code System Technion”, NP-565, Israel Institute of Technology, Haifa, Israel.

Barry, R. F., 1963, “LEOPARD: A Spectrum Dependent Non-spatial Depletion Code” WCAP-3269-26, Westinghouse Electric Corporation.

Bergles, A. E., Rosenow, W. M., 1964, “The determination of forced-convection surface boiling heat transfers”, Transactions of the ASME 86 (series C-J of Heat Transfer), pp. 365-375.

Briemeister, J. F., 2000, “MCNP: A General Monte Carlo N-Particle Transport Code”, Version 4C , LA-13709-M, Los Alamos National Laboratory, Los Alamos.

Fowler, T. B., Vondy, D. R., Cunningham, G. W., 1972, “Nuclear Reactor Core Analisys Code: CITATION”, Oak Ridge National Laboratory, ORNL-TM-2496.

IAEA – TECDOC – 233, 1980, “Research Reactor Core Conversion from Use of High Enriched Uranium to Use Low Enriched Uranium Fuels Handbook”, International Atomic Energy Agency, Vienna, Austria.

Kerr, W., King, J. S., Martin, J. C., Wehe, D. K., 1991, “The Nuclear Ford Reactor Demonstration Project Goes the Evaluation and Analisys of Low Enrichment Fuel – Final Report” , ANL/RERTR/TM-17, Argonne National Laboratory, Argonne.

Klein,S. A., Alvarado, F. L., 1996, “EES - Engineering Equation Solver for Microsoft Windows Operating System”, Version 4.334, Middleton, WI: F-Chart Software.

Torres, W. M., Umbehaun, P. E., Baptista F^o, B. D., Almeida, J. C., Souza, J. A. B., Silva, D. G., 2001, “ Distribuição de Vazão no Núcleo do Reator de Pesquisas IEA-R1”, XVI Brazilian Congress of Mechanical Engeneering, Uberlândia, Minas Gerais, Brazil.

Umbehaun, P. E., 2000, “Metodologia para Análise Termo-Hidráulica de Reatores de Pesquisa Tipo Piscina com Combustível Tipo Placa”, M. Sc. Thesis, IPEN – Instituto de Pesquisas Energéticas e Nucleares.

Whittle, R.H., Forgan, R., 1967, “Correlation for the minima in the pressure drop versus flow rate curves for subcooled water flowing in narrow heated chanels”, Nuclear Engineering and Design, vol 6.