2009 International Nuclear Atlantic Conference - INAC 2009
Rio de Janeiro,RJ, Brazil, September27 to October 2, 2009
ASSOCIAÇÃO BRASILEIRA DE ENERGIA NUCLEAR - ABEN
ISBN: 978-85-99141-03-8

NEUTRONIC, THERMAL-HYDRAULIC AND ACCIDENT ANALYSIS CALCULATIONS FOR AN IRRRADIATION DEVICE TO BE USED IN THE QUALIFICATION PROCESS OF DISPERSION FUELS IN THE IEA-R1 RESEARCH REACTOR

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ABSTRACT

Neutronic, thermal-hydraulics and accident analysis calculations were developed to estimate the safety of an irradiation device placed in the IEA-R1 reactor core. The irradiation device will be used to receive miniplates of U_3O_8-Al e U_3Si_2-Al dispersion fuels, LEU type (19.9% of ^{235}U), with uranium densities of, respectively, 3.0 gU/cm^3 and 4.8gU/cm^3. The fuel miniplates will be irradiated to nominal ^{235}U burnup levels of 50% and 80%, in order to qualify the above high-density dispersion fuels to be used in the Brazilian Multipurpose Reactor, now in the conception phase. For the neutronic calculation, the computer code CITATION was utilized. The computer code FLOW was used to calculate the coolant flow rate in the irradiation device, allowing the determination of the fuel miniplate temperatures with the computer model MTRCR-IEA-R1. A postulated Loss of Coolant Accident (LOCA) was analyzed with the computer codes LOSS and TEMPLOCA, allowing the calculation of the fuel miniplate temperatures after the reactor pool draining. The calculations showed that the irradiation of the fuel miniplates will happen without any adverse consequence in the IEA-R1 reactor.

1. INTRODUCTION

The IEA-R1 reactor of IPEN-CNEN/SP in Brazil is a pool type research reactor cooled and moderated by demineralized water and having Beryllium and Graphite as reflectors. In 1997 the reactor received the operating licensing for 5 MW. Since 1998, IPEN has been producing and qualifying its own U_3O_8-Al and U_3Si_2-Al dispersion fuels. The U_3O_8-Al dispersion fuel is qualified up to a uranium density of 2.3 gU/cm^3 and the U_3Si_2-Al dispersion fuel up to 3.0 gU/cm^3 [1]. The IEA-R1 reactor core is constituted of the fuels above, with low enrichment in ^{235}U (19.9% of U-235). IPEN has no hot cells to provide destructive analysis of the irradiated nuclear fuel. As a consequence, non destructive methods have been used to evaluate irradiation performance of the fuel elements [2]. For fuel qualification, complete fuel elements were irradiated in the IEA-R1 core and the fuel element evaluation has consisted of two items: i) monitoring the fuel performance during the IEA-R1 operation, concerning the following parameters: reactor power, time of operation, neutron flux at the position of each fuel assembly, burnup, inlet and outlet water temperature in core, water pH, water conductivity, chloride content in water, and radiochemistry analysis of reactor water; ii) periodic underwater visual inspection of fuel assemblies and eventual sipping test for fuel element suspect of leakage. Irradiated fuel elements have been visually inspected periodically by an underwater radiation-resistant camera inside the IEA-R1 reactor pool, to verify its integrity and its general plate surface conditions.

Nowadays, IPEN-CNEN/SP is interested in qualifying the above dispersion fuels at higher densities. The uranium densities of 3.0 gU/cm^3 and 4.8 gU/cm^3, respectively, for U_3O_8-Al and U_3Si_2-Al, are the maximal uranium densities qualified in the world for these dispersion fuels. Ten fuel miniplates, five of U_3O_8-Al fuels and five of U_3Si_2-Al fuels, with densities of, respectively, 3.0 gU/cm^3 and 4.8 gU/cm^3, were fabricated at IPEN-CNEN/SP. The miniplates will be put in an irradiation device, with same external dimensions of IEA-R1 fuel elements. The irradiation device will be placed in a peripheral position of the IEA-R1 reactor core. The miniplates will be irradiated to nominal ^{235}U burnup levels of 50% and 80%. To certify miniplate safety irradiation in the IEA-R1 reactor core, neutronic, thermal-hydraulics and accident analysis calculations were developed. For the neutronic calculation, the computer code CITATION was utilized. The computer code FLOW was used to calculate the coolant flow rate in the irradiation device, allowing the determination of the fuel miniplate temperatures with the computer model MTRCR-IEA-R1. A postulated Loss of Coolant Accident (LOCA) was analyzed with the computer codes LOSS and TEMPLOCA, allowing the calculation of the fuel miniplate temperatures after the reactor pool draining. The calculations showed that the irradiation of the fuel miniplates will happen without any adverse consequence in the IEA-R1 reactor.

2. NEUTRONIC, THERMAL-HYDRAULICS AND ACCIDENT ANALYSIS CALCULATIONS FOR THE IRRADIATION DEVICE

A special Miniplate Irradiation Device (MID) was designed for irradiation of the fuel miniplates in the IEA-R1 reactor. The Figure 1 shows the MID.

Figure 1. – Miniplate Irradiation Device – MID.

The MID has the external dimensions of the IEA-R1 fuel element. The miniplates will be allocated in a box with indented bars placed inside the external part of the MID. The Figure 2 shows the transversal section of the MID.

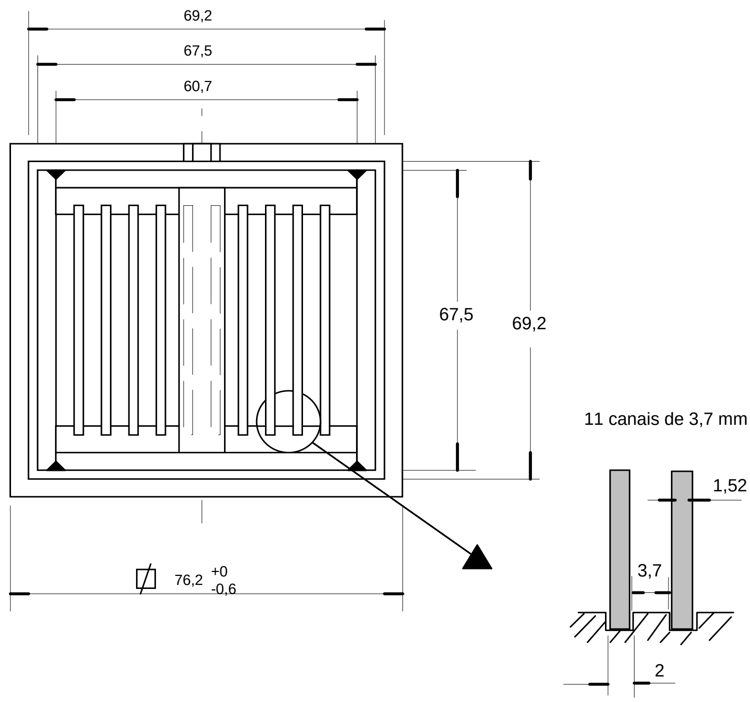

Figure 2. Transversal section of the MID (dimensions in mm).

As can be seen from the Figure 2, up to ten miniplates can be placed in the box with indented bars inside of the MID. The qualification of the U_3O_8-Al and U_3Si_2-Al dispersion fuels with higher ^{235}U density will be made in use, which means that it is based on the irradiation of the dispersion fuel miniplates in the IEA-R1 reactor followed by the use of non-destructive analysis techniques, mainly fuel miniplate visual inspections performed regularly with a radiation-resistant underwater camera.

A new special system [3] was designed for the fuel miniplate swelling determination. The swelling determination will be by means of the fuel miniplate thickness measurement during the irradiation time in the IEA-R1 reactor. The so called “Fuel Miniplate Thickness Measurement System” will be located at the fuel storage area of the IEA-R1 reactor pool. It will be operated from the reactor pool border, allowing the measurement of the fuel miniplate thickness along its surface by electronic probes (LVDT). The results will be collected by instrumentation connected to the probes.

Fuel performance evaluation can be summarized by the fuel miniplate average burnup at the end of its whole irradiation period in the reactor core.

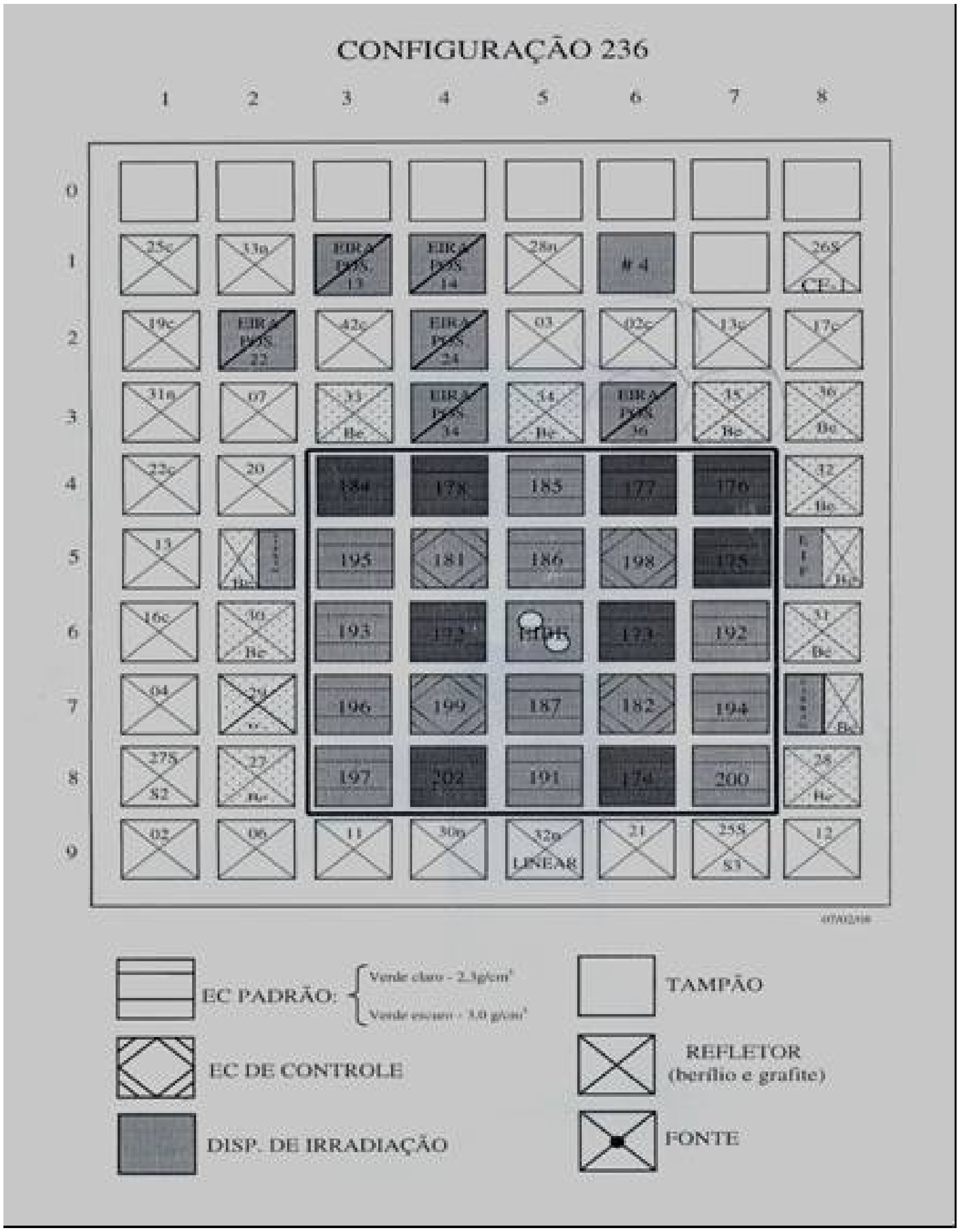

Figura 3. – IEA-R1 reactor core.

2.1 Neutronic calculation

In the neutronic calculation, the computer code CITATION [4] was utilized for the three-dimensional core calculation and for burnup calculation. The cross sections were generated by the computer code TWODB [5]. The calculated radial and axial density curves were utilized as input data for the thermal-hydraulics reactor core and MID analysis. It was utilized the core configuration number 236 of the IEA-R1 reactor, showed in Figure 3, with 24 elements, being 20 standard elements of U_3O_8-Al and U_3Si_2-Al dispersion fuels, four control elements, and one beryllium irradiation element in the central position of the core. In the calculation it was considered that every fuel miniplate was U_3Si_2-Al with an uranium density of 4.8 gU/cm^3. This situation will generate the maximal power density in the miniplates, due the maximal uranium density used.

Three positions in reactor core matrix plate were offered by the reactor operation to place the MID: 1) position 26; 2) position 37; and 3) position 36. In order to define the best position for the miniplate irradiation, calculations of the power density were developed for the three positions. The Tables 1 to 3 present de axial power density results for a reactor power of 5 MW. The miniplate volume used in the calculations was 123.19 cm^3, with an active meat height of 11.8 cm. From the tables can be seen that the maximal power density was obtained for the position 36. This position in the reactor matrix plate was choose for placing the MID which will achieve in this position the higher burnup in a shorter period of time.

Table 1. Miniplate power density calculated for the irradiation device placed in the position 26 of the matrix plate of IEA-R1 reactor core

Power Density$^{(*)}$ (W/cm^3) - U_3Si_2 – 4,8 gU/cm^3			
Elevation	Height (cm)	Region 1	Region 2
1.	0,0	70,880	94,016
2.	1,18	65,780	89,227
3.	2.36	63,504	86,873
4.	3,54	62,247	85,419
5.	4,72	61,419	84,365
6.	5,9	60,825	83,535
7.	7,08	60,456	82,918
8.	8,26	60,482	82,666
9.	9,44	61,427	83,225
10.	10,62	64,909	85,966

$^{(*)}$Volume for the power calculation = 0.522 x 5.0 x 4.0 x 11.8 = 123.19 cm^3.

Table 2. Miniplate power density calculated for the irradiation device placed in the position 37 of the matrix plate of IEA-R1 reactor core

Power Density $^{(*)}$ (W/cm^3) - U_3Si_2 – 4,8 gU/cm^3			
Elevation	Height (cm)	Region 1	Region 2
1.	0,0	96,213	104,62
2.	1,18	89,322	96,989
3.	2.36	86,122	93,424
4.	3,54	84,303	91,397
5.	4,72	83,111	90,072
6.	5,9	82,285	89,159
7.	7,08	81,810	88,638
8.	8,26	81,902	88,749
9.	9,44	83,226	90,223
10.	10,62	87,827	95,290

$^{(*)}$Volume for the power calculation = 0.522 x 5.0 x 4.0 x 11.8 = 123.19 cm^3.

Table 3. Miniplate power density calculated for the irradiation device placed in the position 36 of the matrix plate of IEA-R1 reactor core

Power Density (*) (W/cm^3) - U_3Si_2 – 4,8 gU/cm^3			
Elevation	Height(cm)	Region 1	Region 2
1.	0,0	106,73	115,62
2.	1,18	98,848	106,97
3.	2.36	95,224	102,99
4.	3,54	93,187	100,75
5.	4,72	91,863	99,295
6.	5,9	90,957	98,300
7.	7,08	90,451	97,743
8.	8,26	90,592	97,894
9.	9,44	92,146	99,593
10.	10,62	97,480	105,41

(*)Volume for the power calculation = 0.522 x 5.0 x 4.0 x 11.8 = 123.19 cm^3.

2.2 Thermal-Hydraulics Analysis

A new thermal-hydraulics model MTCR-IAE-R1 [6] was developed in 2000 at IPEN-CNEN/SP in using a commercial program Engineering Equation Solver (EES). The use of this computer model enables the steady-state thermal and hydraulics core analyses of research reactors with MTR fuel elements. The following parameters are calculated along the fuel element channels: fuel meat central temperature (T_c), cladding temperature (T_r), coolant temperature (T_f), the Onset of Nucleate Boiling (ONB) temperature (T_{onb}), the critical heat flux (Departure of Nucleate Boiling-DNB), flow instability and the thermal-hydraulics safety margins MDNBR and FIR. The thermal-hydraulics safety margins MDNBR and FIR are calculated as the ratio between, respectively, the critical heat flux and the heat flux for flow instability and the local heat flux in the fuel plate. Furthermore, the MTCR-IEA-R1 model also utilizes in its calculation the involved uncertainties in the thermal-hydraulics calculation as, for instance, fuel fabrication uncertainties, error in the power density distribution calculation, in the coolant flow distribution in the core, reactor power control deviation, in the coolant flow measures, and in the safety margins for the heat transfer coefficients. The calculated thermal and hydraulics core parameters are compared with the design limits established for MTR fuels: a) cladding temperature < 95°C; 2) safety margin for the onset of nucleate boiling higher than 1.3, or the ONB temperature higher than coolant temperature; 3) safety margin for flow instability higher than 2.0; and 4) safety margin for critical heat flux higher than 2.0.

The placement of the MID in the matrix plate of IEA-R1 reactor will deviate part of the reactor coolant flow to cool the fuel miniplates. The flow in the core of the reactor IEA-R1 is 3400 gpm which provides a flow rate of approximately $20m^3/h$ for fuel element, flow sufficient to cool a standard fuel element. From these data were developed parametric tests with the program MTCR-IEA-R1 shown the need for a minimum flow of $10m^3/h$ in the MID. The simulations have considered the MID with ten identical miniplates, same composition (U_3Si_2-Al), same uranium density (4.8 gU/cm^3), in order to analyze the most critical device configuration.

The simulation results presented in Figure 4 show that no design limit was achieved for the analyzed core. The calculated cladding temperatures are under the value of 95°C, and the coolant temperatures for all coolant flow rates tested are below the ONB temperature, indicating that one-phase flow in the MID. The margins for critical heat flux (MDNBR) and flow instability (FIR) are well above the value of 2.0, admitted as design limit. The maximal fuel meat temperature was 93°C. With these results, it was defined the value of 12,3m^3/h for the coolant flow rate in the MID. This value is sufficient to cool the miniplates and represent only a small deviation of the total coolant flow rate in the reactor core. A special flow restrictor was fabricated in order to maintain this coolant flow rate in the MID (see Figure 1).

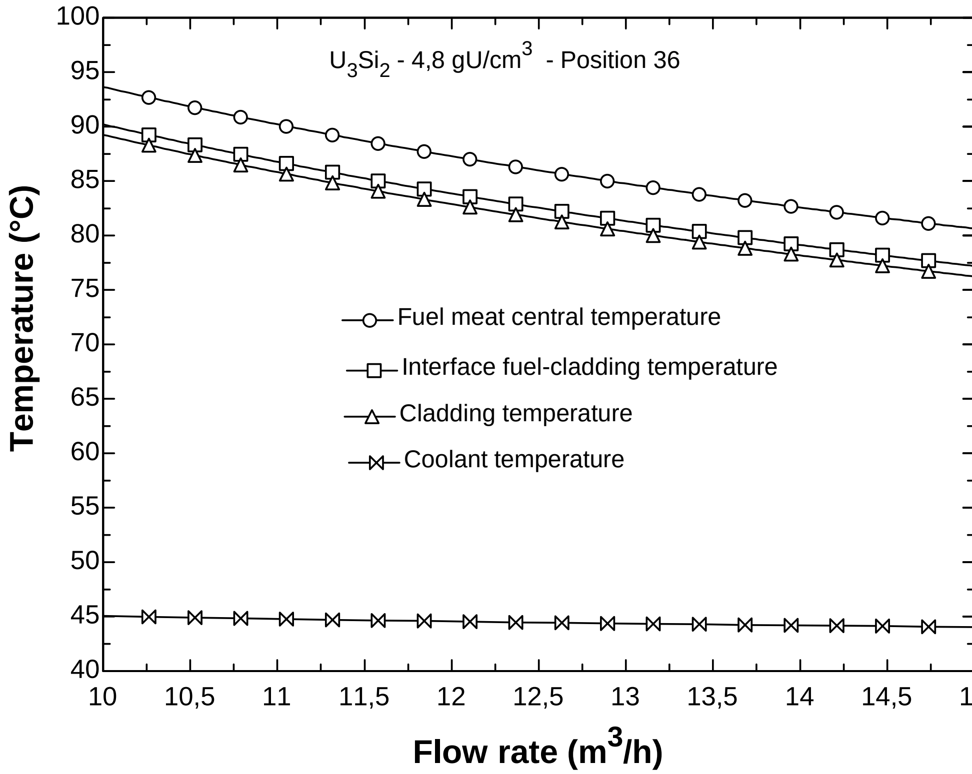

Figura 4. Fuel miniplate and coolant calculated temperatures versus MID coolant flow rate (U_3Si_2 – 4.8 gU/cm^3 – Position 36 of the matrix plate of IEA-R1 reactor.

2.3 Accident Analysis

The Loss of Coolant Accidents (LOCA) in pool type research reactors are normally considered as limiting in the licensing process. For the IEA-R1 research reactor, a special Emergency Cooling System (ECS) was constructed in order to avoid the core melting during a postulated Primary Coolant Boundary Rupture. With two redundant systems with passive action, the ECS will be able to cool the reactor core after a reactor pool draining. However, the action of this system is limited to the boundaries of the reactor core and the ECS is not able to cool the MID which is placed in a reactor core peripheral position. Two computer codes LOSS and TEMPLOCA [7] were used to calculate the temperatures achieved in the fuel miniplates during the core draining. The computer code LOSS determines the time to drain the reactor pool down to the level of the bottom of the core, and the computer code

TEMPLOCA can calculates the maximal temperature achieved in the fuel miniplate during this transient. The Figure 5 shows that about 7.5 min is necessary to drain the reactor pool during a postulated Primary Coolant Boundary Rupture accident. After the pool draining, Figure 6 shows that the maximal fuel plate temperature achieved was 180 ºC, below the blistering temperature, which is the fuel temperature design limit [8]. At the blistering temperature the fuel minplate will swell due the fission gases released in the fuel and can close the fuel miniplate cooling channels with temperature increasing up to miniplate melting. The value of blistering temperature for dispersion fuels can vary between 350°C e 600 °C, depending on dispersion fuel type, fuel enrichment and burnup achieved.

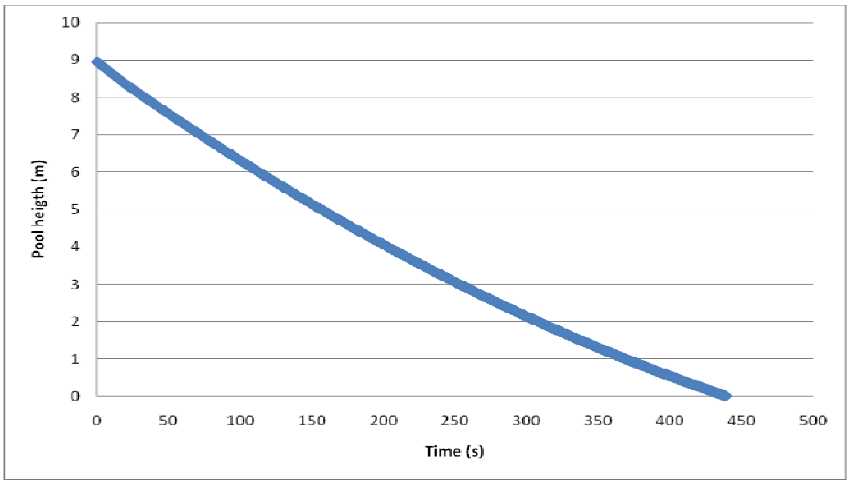

Figura 7. Time to drain the reactor pool down to the level of the bottom of the core during a Primary Coolant Boundary Rupture postulated accident.

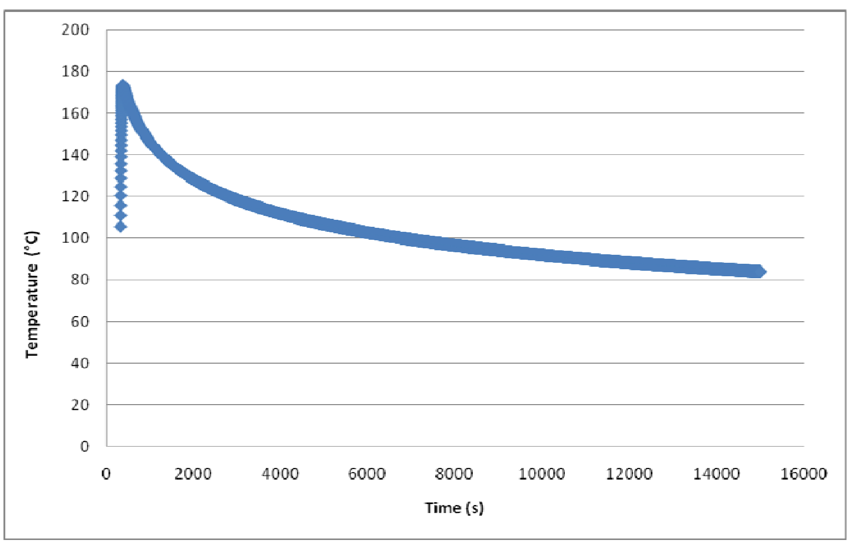

Figura 8. Minplates temperatures achieved after core draining in a postulated Primary Coolant Boundary Rupture accident.

3. CONCLUSIONS

The result of neutron calculations showed that the best position for the irradiation in the reactor core is the 36, for having the highest power density, and the inclusion of MID in the reactor core does not affect the operation of the same, since the change in reactivity is irrelevant. Through the thermal-hydraulics calculations it was determined a minimum flow for cooling the fuel miniplates diverting a small fraction of the flow of the reactor core without affecting the cooling of the core. From the analysis of accidents concluded that there is any damage to miniplates in the case of a postulated Primary Coolant Boundary Rupture.

ACKNOWLEDGMENTS

The authors are grateful for financial support from the Fundação de Amparo à Pesquisa do Estado de São Paulo (FAPESP).

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