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URANIUM DENSITY REDUCTION ON FUEL ELEMENT SIDE PLATES ASSESSMENT

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ABSTRACT

During operation of IEA-R1 research reactor, located at Instituto de Pesquisas Energéticas e Nucleares, IPEN – CNEN/SP, an abnormal oxidation on some fuel elements was noted. It was also verified, among the possible causes of the problem, that the most likely one was insufficient cooling of the elements in the core.
One of the propositions to solve or minimize the problem is to reduce uranium density on fuel elements side plates. In this paper, the influence of this change on neutronic and thermal hydraulic parameters for IEA-R1 reactor is verified by simulations with the codes HAMMER and CITATION.
Results are presented and discussed.

1. INTRODUCTION

1.1 Historical

Reactor IEA-R1 operational experience showed that after the power increase from 2MW to 5MW and the reduction core, there was a growth on side plate oxidation in some fuel elements.

It is known that some factors may cause this behavior such as water quality, cladding temperature and cladding composition, among others.

Several studies [1, 2, 3] and visual inspections were conducted by Nuclear Engineering Center, CEN, as an attempt to identify the possible causes of oxidation.

Once the water chemical treatment system was recently substituted and the established limits were verified and they are completely granted, certifying that the quality of the water fits reactor operation demand.

Regarding cladding composition, the supplied materials are in accordance with technical specifications; therefore a more detailed investigation is not necessary.

Another hypothesis, the most probable one, is the insufficient fuel element cooling.

As an attempt to a better evaluation of core cooling conditions, a dummy fuel element was projected and built to measure fuel element total flow and cooling flow for each inner canal. Through experimental measures, the verified flow was below the expected value. As a consequence, a research to identify the reasons of fuel element flow began. The results of this research showed that there were flow measure problems, such as calibration and measure errors, and flow deviation through plugs and irradiators above the estimated. With mentioned problems proper corrections, it was achieved an increase at about 30% on fuel element effective inner flow value [2].

Nevertheless, oxidation problem endured, though thermal-hydraulics calculation did not indicate high temperatures presence.

From this moment on, the suspects on the cause of the problem passed to low flow between fuel elements, once they are open canals and allow transversal flow as presented and discussed in Umbehaun [4].

Uranium density reduction on fuel element side plates in order to reduce their generated power was a proposed solution for this problem. This uranium density reduction is observed in other reactors, such as reactor RECH-1, from Chile [5]. Though the purpose of this reduction is not defined on available bibliography, it seems to happen for the same reason as in reactor IEA-R1.

In the case of ascending outflow/outlet in the core, some other reactors adopt an enclosure system (similar to a chimney) or even the use of a type of guide, such as in German reactor FRG1 [6]. As in the example above, the purpose of this guide is not defined on available literature; however it is assumed that the main purpose of these changes is to assure the appropriate flow in canals between fuel elements.

Effective change of uranium density reduction has already taken place in some fuel elements at IEA-R1 and burnup analysis and maintenance or safety reactor operation parameters became necessary.

2 REACTOR CORE

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 [7]. The IEA-R1 reactor core, Figure 1, is constituted of the fuels above, with low enrichment in ^{235}U (19.9% of U-235). Nowadays, reactor IEA-R1 works at 4MW power.

The reactor core studied in this paper is an ideal one: all fuel elements are new, with only one type of fuel (U3Si2-Al) and they have not been burnt yet. Fuel elements are arranged in a square 5 x 5 and it was necessary to simulate two situations: in the first one, all plates of all fuel elements present 1,2 g/cm^3 uranium density, whereas for the second simulation, external fuel elements side plates had a reduction on uranium density to 1,0 g/cm^3.

This initial reactor operation condition was chosen to allow a more precise analysis of neutronic and thermal-hydraulics parameters of the reactor and also to obtain high reliability results when analyzing the impact of reactor core configuration change.

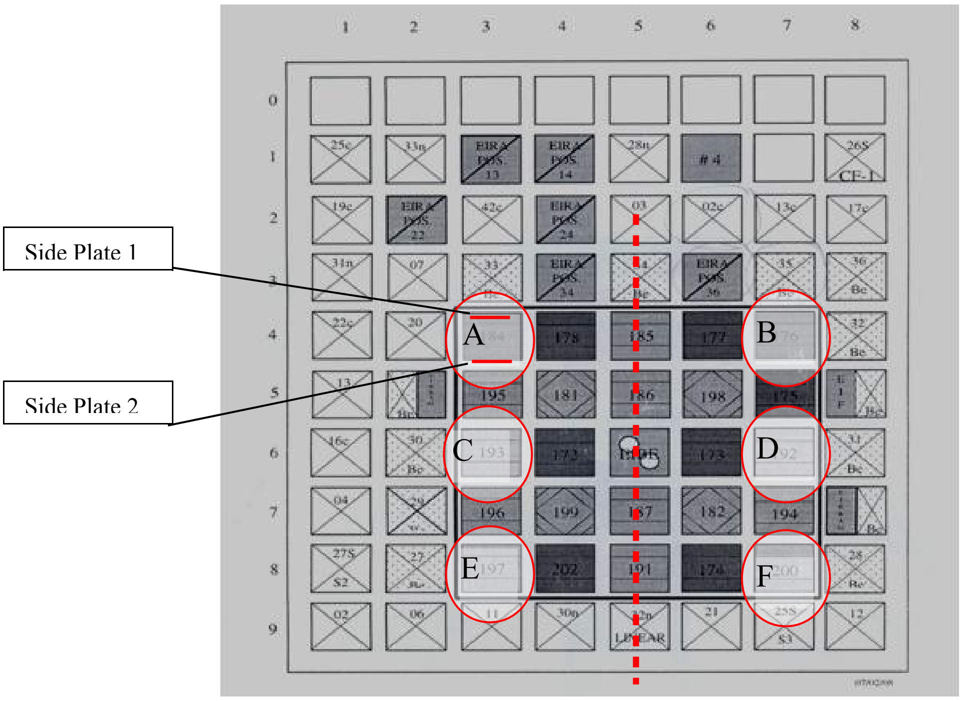

Figure 1. Reactor configuration, elements position, side plates and symmetry axis.

3 NEUTRONIC ANALYSIS

Computer codes HAMMER and CITATION were used for neutronic calculation. The cross sections were generated by the computer code HAMMER [9]. Computer code CITATION [8] was used for the three-dimensional core calculation and also for burnup calculation. The calculated radial and axial density curves were used as input data for the thermal-hydraulics reactor core analysis.

3.1 Computer Code HAMMER

Computer code HAMMER calculates cross sections for materials in reactor core, once the user provides each material initial concentration at the beginning of reactor operation. These values are obtained from fuel element and control rod technical specifications.

Cross section values HAMMER output are CITATION input to calculate neutronic parameters.

3.2 Computer Code CITATION

Computer code CITATION calculates the following neutronic parameters effective-k, neutron flux and power density, for this paper. For such results, CITATION input is materials cross sections, calculated using computer code HAMMER; then before calculating neutron fluxes and power density, the value of effective-k must be verified. For reactor IEA-R1, this parameter maximum value is 1,07.

3.3 Neutronic results

Figure 1 shows the six fuel elements studied for this paper; that were divided into three pairs, according to their symmetry in relation to an imaginary axis, and the side plates analyzed, named 1 and 2; element A was highlighted as an example of the procedure adopted for all six fuel elements: the two more external side plates in each fuel element, once they were the modified ones. Two simulations took place with computer code CITATION: the first one named as 'Original', with 1,2 g/cm^3 uranium density in all fuel elements plates and the second one, named as 'New', with uranium density reduction from 1,2 g/cm^3 to 1,0 g/cm^3. Results from the comparisons between these two situations are shown and commented below, Figure 2 to Figure 7.

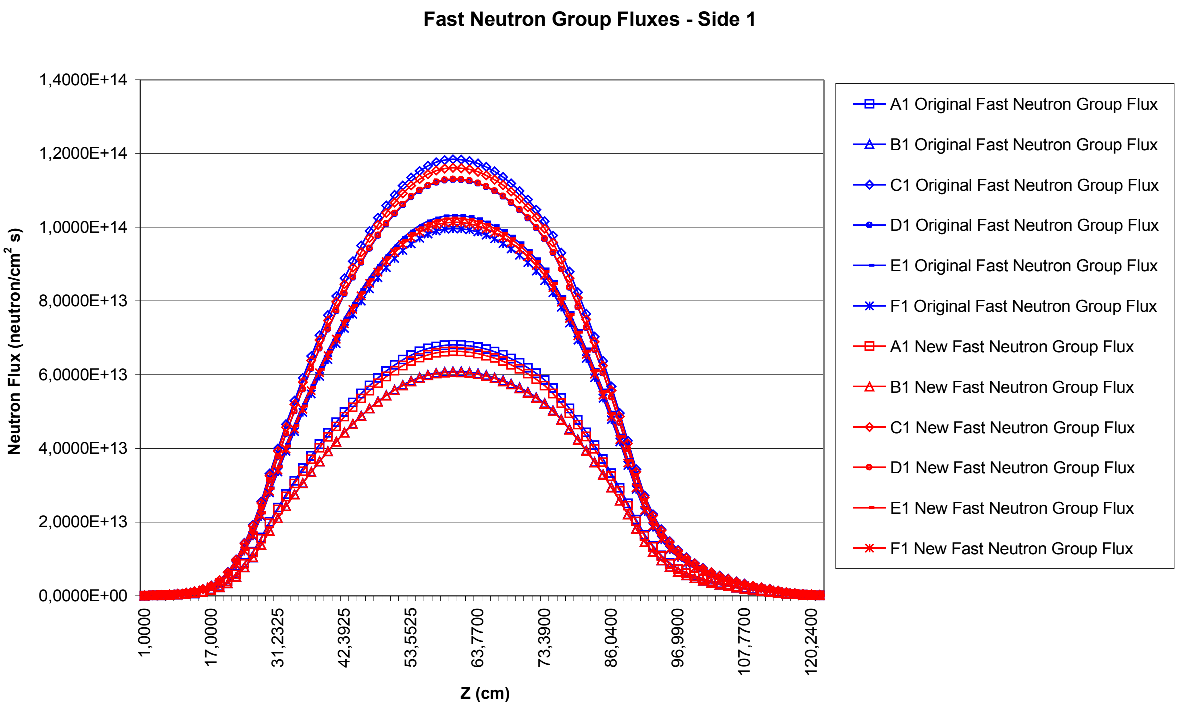

Figure 2. Original and new fast neutron group fluxes for fuel element side plates 1.

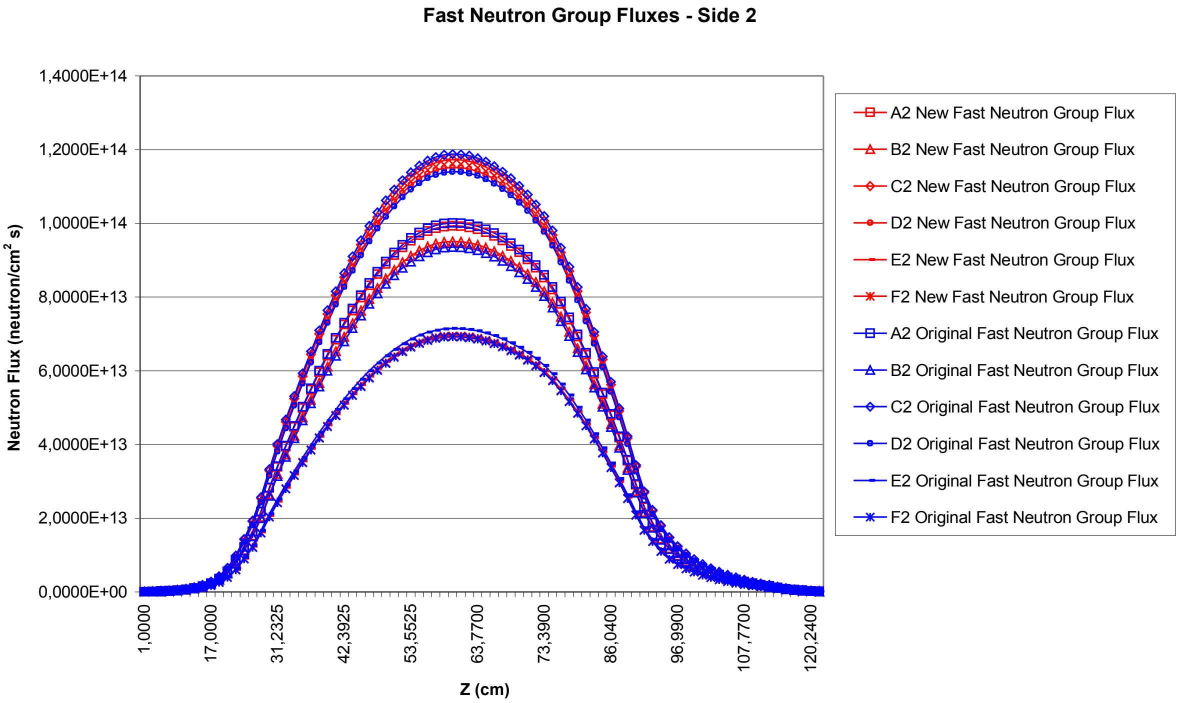

Figure 3. Original and new fast neutron group fluxes for fuel element side plates 2.

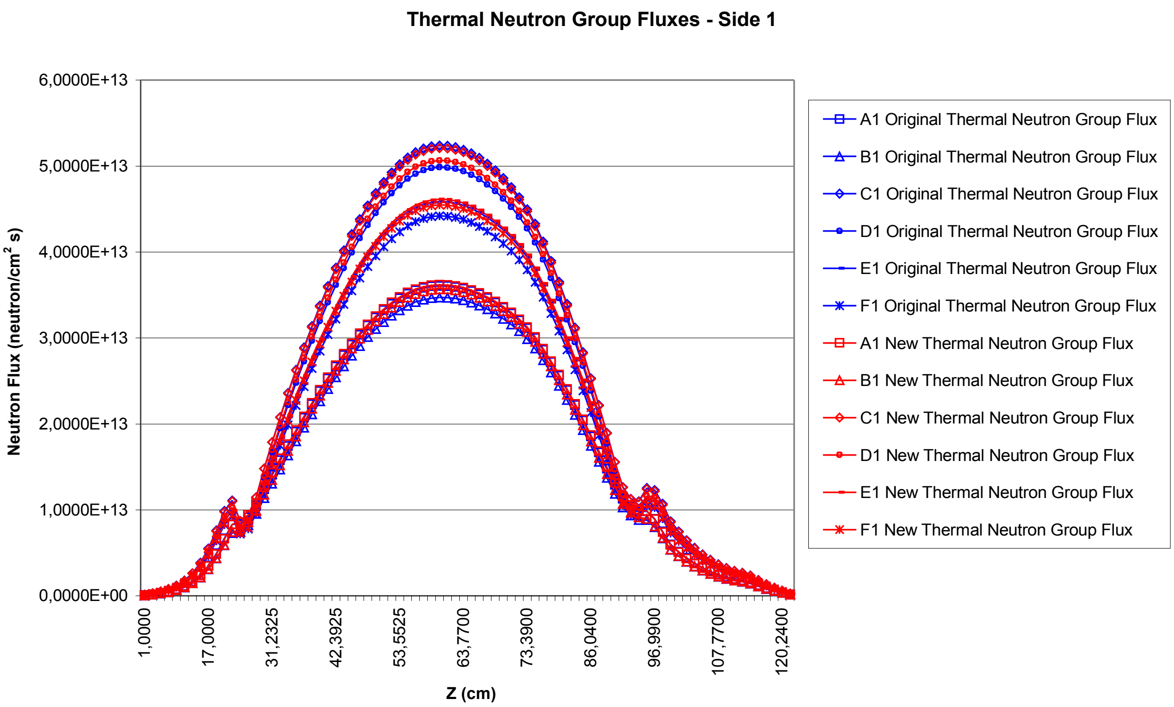

Figure 4. Original and new thermal neutron group fluxes for fuel element side plates 1.

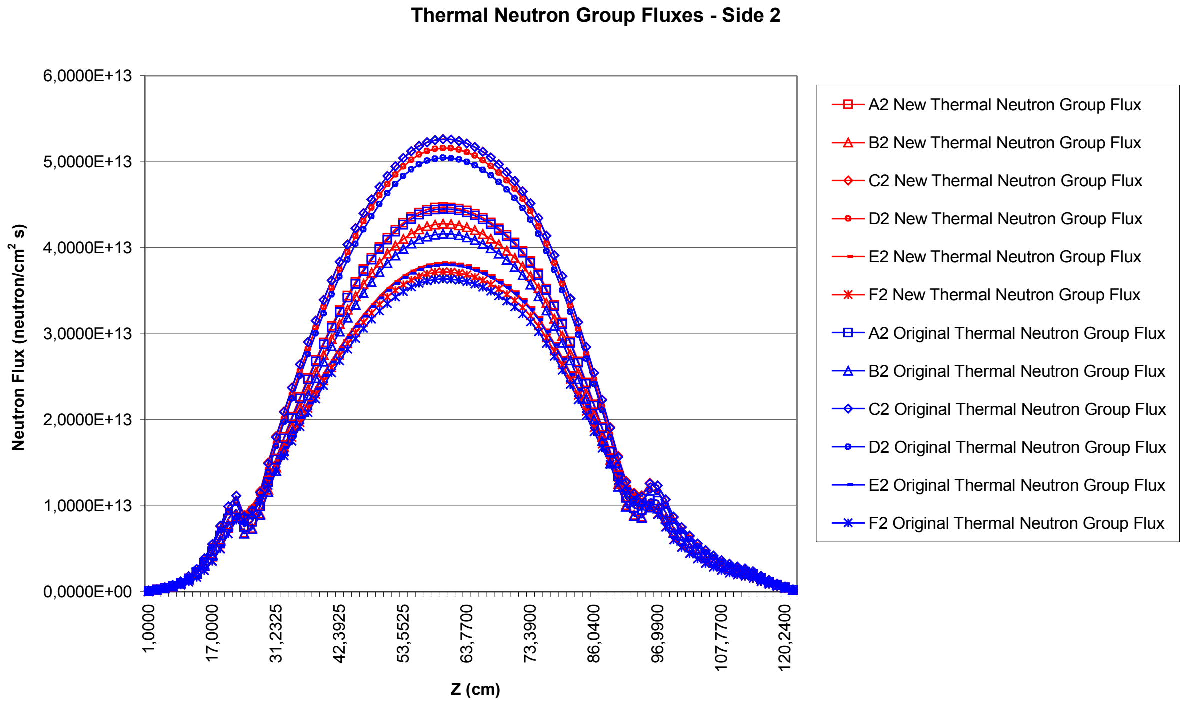

Figure 5. Original and new thermal neutron group fluxes for fuel element side plates 2.

Figure 6. Original and new power densities for fuel element side plates 1.

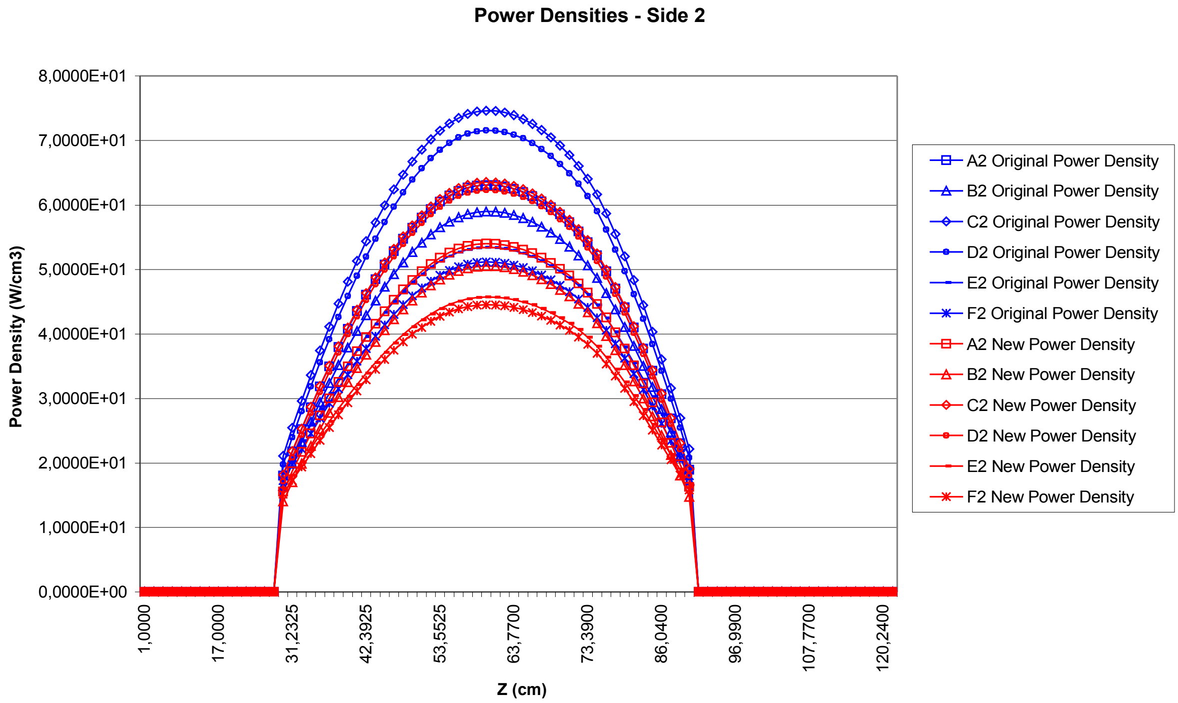

Figure 7. Original and new power densities for fuel element side plates 2.

The behavior presented for fast and thermal neutron groups in both situations is according to the expected: fuel element side plates located at the center of the core (fuel elements C and D) presented bigger neutron fluxes than other elements side plates, as shown in Figure 2 for fast neutron group flux on the six fuel element side plates 1; in Figure 3 for fast neutron group flux on the six fuel element side plates 2; in Figure 4 for thermal neutron group flux on the six fuel element side plates 1; in Figure 5 for thermal neutron group flux on the six fuel element side plates 2. The other two fuel element pairs, A and B and C and D, presented the same magnitude for their neutron fluxes, behavior highlighted by their almost superposed curves, either for original and new reactor operational conditions.

The expected behavior for power density was noted in Figure 6 that shows power density for the six fuel element side plates 1and Figure 7, which presents power density for the six fuel element side plates 2; fuel elements C and D showed bigger power density than the other four studied elements. Fuel elements A and B showed power density at the same magnitude and this behavior was also noticed for fuel elements pair E and F.

Reactor core criticality – parameter k – was not significantly changed with fuel element side plates modification. To start reactor operation, k must be as near to unity as possible, which means the system is critic, but a little above this value to counterbalance fission products generated during burnup. In the case of reactor IEA-R1, k value must be at most 1,07. Original k value is 1,03929 and modified core k value is 1,0377022, which means that even with side plates uranium density reduction, initial criticality is almost the same in both situations. This result is expected, once side plates uranium density reduction is small: from 1,2 g/cm^3 to 1,0 g/cm^3 and it took place only in some fuel elements side plates, not in all core elements.

Tables 1 and 2 show both fast end thermal neutron group fluxes and power density value peaks for original and new reactor operational conditions for side plates 1 and 2 respectively.

Table 1. Neutron groups fluxes and power density peaks for Side Plate 1

Element	Original Fast Neutron Group	New Fast Neutron Group	Original Thermal Neutron Group	New Thermal Neutron Group	Original Power Density	New Power Density.
A	6,8089E+13	6,6321E+13	3,6057E+13	3,6191E+13	50,692	43,3
B	6,0906E+13	6,0682E+13	3,4716E+13	3,5672E+13	48,650	41,615
C	1,1843E+14	1,1612E+14	5,2382E+13	5,2097E+13	74,296	62,917
D	1,1290E+14	1,1309E+14	4,9855E+13	5,0652E+13	70,718	61,177
E	1,0321E+14	1,0248E+14	4,5860E+13	4,6132E+13	65,031	55,703
F	9,9641E+13	1,0136E+14	4,4195E+13	4,5472E+13	62,675	54,916

Table 2. Neutron groups fluxes and power density peaks for Side Plate 2

Element	Original Fast Neutron	New Fast Neutron	Original Thermal Neutron	New Thermal Neutron Group	Original Power Density	New Power Density
A	1,0014E14	9,9347E+13	4,4504E13	4,4733E+13	63,107	54,013
B	9,3594E+13	9,5121E+13	4,1609E+13	4,2784E+13	59,000	50,538
C	1,1882E+14	1,1731E+14	5,2611E+13	5,2624E+13	74,617	63,554
D	1,1402E+14	1,1522E+14	5,0451E+13	5,1602E+13	71,556	62,324
E	7,1517E+13	6,9813E+13	3,8015E+13	3,8235E+13	53,436	45,738
F	6,9254E+13	6,9464E+13	3,6378E+13	3,7193E+13	51,163	44,536

4 THERMAL HYDRAULICS ANALYSIS

Next step on this study is core thermal hydraulics analysis, to calculate and analyze thermal hydraulics parameters change with core reactor modification. A thermal-hydraulics model MTCR-IAE-R1 [4] was developed in 2000 at IPEN-CNEN/SP 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 (Tc), cladding temperature (Tr), coolant temperature (Tf), the Onset of Nucleate Boiling (ONB) temperature (Tonb), the critical heat flux (Departure of Nucleate Boiling-DNB), flow instability and the thermal-hydraulics safety margins. The thermal-hydraulics safety margins are calculated as the ratio between 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 uses 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 will be 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.

5 CONCLUSIONS

Obtained results are according to the expected behavior: with uranium density reduction on fuel element side plate neutron fluxes for both fast and thermal groups decreased, as well as power density. Next step expected results, from thermal-hydraulics analysis, are decreased temperatures on external modified side plate fuel elements in comparison with the original situation. From this result, a reduction on side plate oxidation is also expected. Uranium density reduction causes also a decrease on reactor criticality (effective-k), as discussed on item 3.3 ('Neutronic Results'), which reduces reactor refueling period and, consequently, an increase on fuel consumption to an equal generated power.

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