BJRS

BRAZILIAN JOURNAL
OF
RADIATION SCIENCES
09-01 (2021) 01-17

SBPR
Sociedade Brasileira de Proteção Radiológica

Comparison of computer programs to analyze the irradiation performance of U-Mo monolithic fuel plates and UO_2 cylindrical fuel rods in power reactors.

Silva[1] A. L. C., Silva[2] A. T.

Diretoria de Desenvolvimento Nuclear da Marinha, São Paulo-SP, Brazil
candido.andre@marinha.mil.br

Instituto de Pesquisas Energéticas e Nucleares, São Paulo-SP, Brazil
teixeira@ipen.br

ABSTRACT

The aim of this work is to present a comparative analysis in terms of the irradiation performance of cylindrical uranium dioxide fuel rods and monolithic uranium molybdenum fuel plates in pressurized light water reactors. To analyze the irradiation performance of monolithic uranium molybdenum fuel plates when subjected to steady state operating conditions in light water pressurized reactors, the computer program PADPLAC-UMo was used, which performs thermal and mechanical analysis of the fuel taking into account the physical , chemicals and irradiation effects to which this fuel is subjected. For the analysis of the uranium dioxide fuel rods, the code FRAPCON was used, which is an analytical tool that verifies the irradiation performance of fuel rods of pressurized light water reactor, when the power variations and the boundary conditions are slow enough for the term permanent regime to be applied. The analysis for a small nuclear power reactor, despite the higher power density applied to the fuel plate in relation to the fuel rod, showed that the fuel plates have lower temperatures and lower fission gas releases throughout the analyzed power history, allowing the use of a more compact reactor core without exceeding the design limits imposed on nuclear fuel.

Key words: nuclear fuel performance, computational simulation, power reactor.

ISSN: 2319-0612
Accepted: 2021-03-08

1. **INTRODUCTION**

Many studies have been carried out in the nuclear area to improve the operational procedures of the nuclear plants in order to promote the reduction of operating costs through the optimization of the nuclear fuel cycle.

Nowadays, when addressing the increase in the consumption of electric energy per capita by society, the use of nuclear energy becomes predominant. However, to make its use feasible, this type of energy production must have competitive characteristics when compared to other options for electricity generation. In this way, the operation of nuclear power plants needs to be optimized, always taking into account aspects related to operational safety with a view to protecting people, the environment and operators.

In this context, the international scientific and technological communities have been developing new classes of reactors that are more sustainable, safer and more efficient, with the objective of providing an improvement in several applications, such as energy production, radioisotope production or even naval propulsion.

For the area of research reactors, radiopharmaceutical production and material testing, the Reduced Enrichment for Research and Test Reactors (RERTR) [1], [2], [3], [4] and [5] which aims to develop new technologies to enable the conversion of research reactors that use fuel with highly enriched uranium to reactors that can operate with low enrichment fuel. From this initiative, new nuclear fuel technologies were researched and developed, including monolithic uranium-molybdenum plates, which will be analyzed in this work.

To enable and accelerate the development and implementation of new fuels for nuclear use, computer programs are used that simulate the behavior of these fuels in the various operating conditions in the reactor. The whole process of development and validation of a new type of nuclear fuel is extremely complex, multidisciplinary and needs adequate planning to perform the necessary research and experimental activities.

In this case, the main interest of this article is to make a comparison with computer programs (codes) of the behavior under irradiation of monolithic uranium-molybdenum fuel plates versus cylindrical uranium dioxide rods, according to the the particularities of each fuel in small reactors.

2. MATERIALS AND METHODS

Every computer program (code) that can be used in the process of simulating and analyzing the performance of nuclear fuel under different operating conditions, must have some fundamental characteristics, regardless of the type of nuclear fuel to be analyzed. A computer program must provide: a) the possibility of inserting several increments of time and power profiles in order to accurately portray the different operational conditions to which the nuclear fuel will be subjected during the irradiation (useful life); b) carrying out a thermal analysis of the fuel and the cladding and a mechanical analysis of the fuel-cladding assembly; c) and determination of the influence of phenomena resulting from irradiation conditions, such as the release of fission gases, and their impacts on the behavior of the fuel-cladding assembly.

2.1. Computational programs (codes) used: FRAPCON 3.5 [6] AND PADPLAC-UMo [7]

The FRAPCON is a tool that analyzes the behavior under irradiation of a fuel rod for light water reactors (LWR), when the variations in power and the boundary conditions are slow enough for the term permanent regime to be applied. This includes situations such as long periods at constant power and slow ramps of power that are typical of the normal operation of a nuclear reactor, analyzing over time, the influence of the most significant phenomena that affect the performance of a fuel rod during irradiation, such as the temperature in the fuel and the cladding, swelling and densification of the fuel due to irradiation, fission gas release, internal pressure of the fuel rod, stresses in the cladding and oxidation of the cladding.

The PADPLAC-UMo is a computer program that runs performance analysis of uranium-molybdenum monolithic fuel plates that operate under steady-state conditions in pressurized light water reactors. The code allows the insertion in the program of all the input variables necessary and required for the execution of the simulations. Subsequently, all physical and mathematical models developed to analyze the performance of the fuel plate are calculated and verified, involving the selection of a time increment and a power profile related to that time increment. In this time increase, the temperatures in the fuel center, at the fuel ends and in the cladding are checked, the deformations arising from the stresses applied to the fuel and the cladding as well as the conductance at the fuel-cladding interface. The amount of fission gases released is also estimated and there is a gap between the outer surface of the fuel and the inner surface of the liner, and the total gas pressure in the fuel compartment is measured. This is sufficient to verify the occurrence of deformations in the cladding that will promote new changes in the conditions of the fuel-cladding interface and, therefore, changes in the conductance values in this interface. The system provides a list of the main results obtained in the simulation process, such as temperatures in the central region and at the edge of the fuel, temperatures on the inside and outside of the cladding, coolant temperatures, production of fission gases and helium gas, as well as the average burn.

In this present work, the FRAPCON will be used to analyze the cylindrical rods and the PADPLAC-UMo will be used to analyze the monolithic plates.

2.2. Presentation of the analysis case - reactor with 58MW nominal power

For the execution of this case, was considered the use of a monolithic uranium-molybdenum fuel plate with the dimensions provided in Table 1.

Table 1: Main dimensions of the fuel plate.

Description	**Values**
Active fuel plate width (m)	0.09
Active fuel plate height (m)	1.00
Cooling channel width (m)	0.09
Fuel plate cladding thickness (m)	0.0002
Fuel plate core thickness (m)	0.002
Transverse area of a cooling channel (cm²)	2.70
Cross-sectional area at the heart of the fuel plate (cm²)	1.80
Number of fuel elements in the reactor (units)	21
Number of fuel plates per fuel element (units)	72
Total number of fuel plates in the reactor core (units)	1512

The dimensions of the fuel plate used in these tests were adopted in order to be similar to the geometric characteristics of monolithic uranium-molybdenum plates used in some of the tests of the RERTR program.

With regard to the thermal hydraulic characteristics adopted for the execution of these tests, the same values established for the first core (rod type) of a small reactor were taken [8]. These values are provided in Table 2.

Table 2: Thermal hydraulic characteristics (fuel plates)

Description	Values
Total rated reactor power (MW)	58
Power generated on the fuel plates (MW)	56.6
Total number of plates in the core (units)	1512
Total power generated on the fuel plate (kW)	38.36
Nominal pressure in the primary system (bar)	130
Mass flow of reactor design (kg/s)	427.6
Coolant inlet temperature (°C)	265.0
Cooling channel thickness (mm)	3
Coolant speed in the cooling channel (cm/s)	146.9

For the idealization of the fuel rod reactor used for comparison in this analysis, was elaborated a reactor with the same volumetric characteristics of the fuel plate reactor, thus providing the same power density for the correct and valid comparison between these two types of reactors . In this case, for the simulation with the FRAPCON code, 21 fuel elements were used, constituted in its unit by 236 rods in a 16x16 matrix, considering the existence of control rods and burnable poison rods, in order to total a set of 4956 fuel rods.

In the simulation performed, the following parameters were adopted: rods with UO_2 cylindrical fuel rods, Zyrcaloy-4 cladding, as well as for all spacer grids and all guide tubes; and total nominal reactor power (MW) = 58 MW, equal to the proposal of the plate type reactor.

Tables 3 and 4 show the characteristics of the fuel rod and the thermo hydraulic conditions for the reactor considered in the analysis.

The thermal models of the FRAPCON code are based on steady state equations and conditions and calculate only the radial heat flow. Similarly, gas releases models are based on steady state data and slow power ramps and do not reflect release rates for rapid power changes. Therefore, variations in the rate of linear heat generation should not exceed 1.5 kW/ft (4.92 kW/m) between time intervals and time intervals should be 0.1 day (2.4 h - 144 min), but maximum of 50 days. It is also necessary to respect a model of gas release and thermal conduction that allow the code to model the behavior of fuel rods up to 62 or 65 [GWd/MTU or MWd/kgU] burns.

Table 3: Fuel rod characteristics

Description	**Values**
claddind outside diameter (mm)	9.70
cladding inside diameter (mm)	8.43
rod outside diameter (mm)	8.25
rod height (mm)	9.91
Chamfer and concavity	Sim
Effective height of rod column (mm)	1000
Enrichment (% U235)	4.3
Cladding thickness (mm)	0.635
Gap thickness (mm)	0.089
Filling gas pressure (MPa)	2.76 (He)
Distance between axes of rods (mm)	12.7
% of theoretical density of UO_2	94
Length of plenum (mm)	85
Number of rods per fuel element	236
Number of fuel elements	21
Total number of fuel rods	4956

Table 4: Thermal hydraulic characteristics (fuel rods)

Description	**Values**
Total rated reactor power (MW)	58
Power generated on the fuel plates (MW)	56.6
Total number of rods in the core (236 x 21) (units)	4956
Total power generated in each fuel rod (kW)	11.70
Nominal pressure in the primary system (bar)	130
Mass flow of reactor design (kg/s)	427.6
Coolant inlet temperature (°C)	265.0
Coolant speed in the cooling channel (cm/s)	146.9

The thermal models of the FRAPCON code are based on steady state equations and conditions and calculate only the radial heat flow. Similarly, gas releases models are based on steady state data and slow power ramps and do not reflect release rates for rapid power changes. Therefore, variations in the rate of linear heat generation should not exceed 1.5 kW/ft (4.92 kW/m) between time intervals and time intervals should be 0.1 day (2.4 h - 144 min), but maximum of 50 days. It is also necessary to respect a model of gas release and thermal conduction that allow the code to model the behavior of fuel rods up to 62 or 65 [GWd/MTU or MWd/kgU] burns.

In this way, respecting the limitations of the code, the following premises were adopted: irradiation period of 2190 days (6 years) envisioning the maximum burn limit of FRAPCON being 62 or 65 GWd/MTU; 45 time increments were adopted, divided as follows: 0.2, 50, 100, 150, 200, 250, 300, 350, 400, 450, 500, 550, 600, 650, 700, 750, 800, 850, 900 , 950, 1000, 1050, 1100, 1150, 1200, 1250, 1300, 1350, 1400, 1450.0, 1500.0, 1550, 1600, 1650, 1700, 1750, 1800, 1850, 1900, 1950, 2000, 2050, 2100, 2150 and 2190. With regard to the power supplied by the fuel rod (11.70 kW), a progressive increase in this power over time was considered and respecting the variations in the linear heat generation rate of a maximum of 1.5 kW/ft (4.92 kW/m), see Table 5.

As for the power supplied by the fuel plate (38.36 kW), the same progressive increase in power was considered over time, respecting variations in the linear heat generation rate of a maximum of 1.5 kW/ft (4.92 kW/m), that in the case of having an effective height of 1 m, we must adopt the limit of 4.92 kW, also seen in Table 5.

Table 5: Power delivered over time

Time steps	Plate Power (KW/m)	Rod Power (KW/m)
1st to 5th time step	28.0	8.53
6th to 10th time step	30.0	9.15
11th to 15th time step	32.0	9.77
16th to 20th time step	34.0	10.36
21st to 25th time step	35.0	10.66
26th to 30th time step	36.0	10.99
31st to 35th time step	37.0	11.28
36th to 45th time step	38.36	11.70

2.3. Design basis

Regarding the reactor with cylindrical fuel rods, the integrity of each rod is guaranteed through the adoption of limits imposed on active loads and defined based on the design criteria. Design boundaries and design criteria form the basis for design. Therefore, it is possible to highlight some criteria and design limits that will be analyzed:

- The maximum temperature in the fuel center must not exceed the fuel melting temperature (assumed limit ≤ 2500 ºC limit for UO_2);
- The maximum internal pressure on the rod must be less than the nominal coolant pressure;
- Stress criterion in the cladding: In the design of the cladding, formed by zircaloy (zirconium alloy), the KWU criterion is adopted. For analysis in the present article, the limit of elastic deformation or flow limit of the material constituting the cladding was verified as a function of the imposed and compressive stresses.
- The maximum equivalent deformation of the cladding is < 1%.

For the reactor with monolithic fuel plates, it can be said that, as in the rods, the integrity of each plate is guaranteed through the adoption of the design bases, which are formed by the limits and design criteria. In this way, it is possible to highlight the main criterion and project limit that will be analyzed:

- The fuel temperature of 420° C must be considered as an operational limit, since, from this point on, the release of fission gases can lead to the appearance of a gap between the fuel and the liner surfaces, compromising the fuel plate operation.

3. RESULTS AND DISCUSSION

The computer simulations aimed to analyze the performance of cylindrical uranium dioxide rods versus monolithic uranium-molybdenum fuel plates, respecting the particularity of each type of fuel.

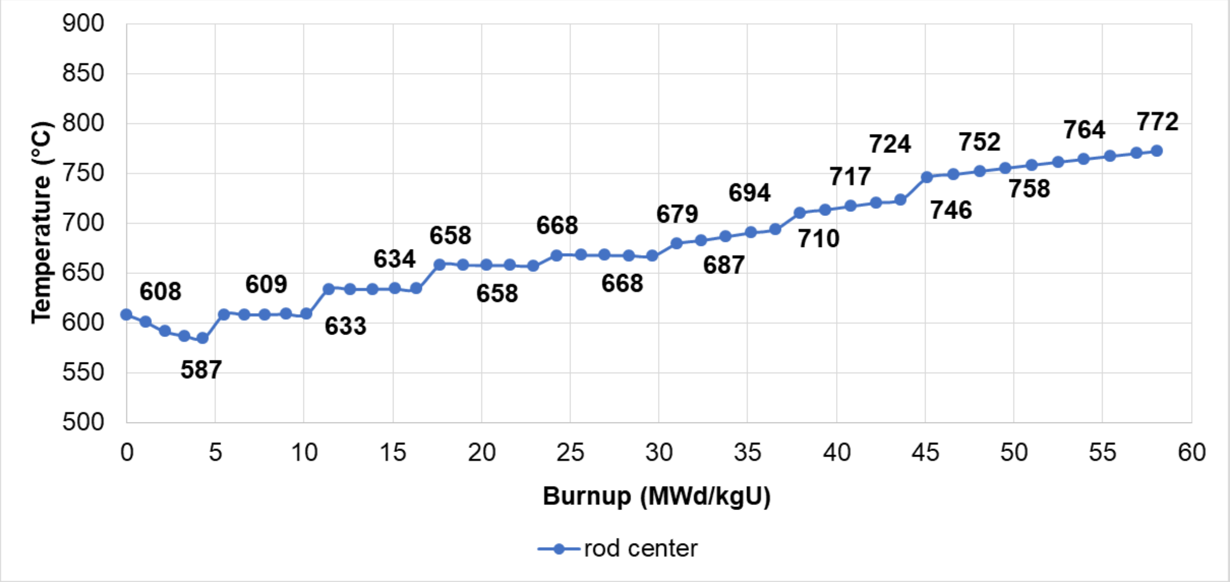
Thus, the items made include: analysis of central fuel temperatures; analysis of the temperatures in the cladding and checking the temperatures in the coolant.

The main objective of the design of a nuclear fuel (rod or plate) is to ensure that under the permitted operating conditions there will be no systematic failure due to the loads to which it is subjected. The operating conditions for design purposes are: 1) normal operation, 2) abnormal conditions, 3) emergency conditions and 4) limiting fault conditions. In this article, were analyzed only data related to the reactor's normal operating condition.

3.1. Fuel center temperature

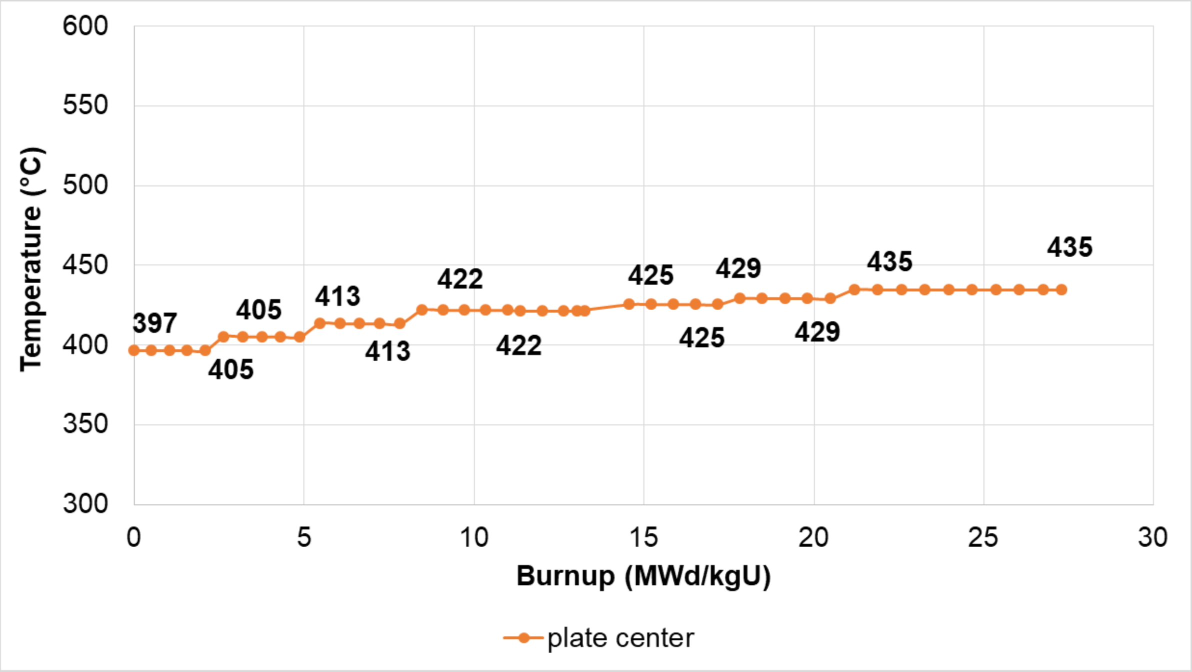
After the proper use of the programs with the insertion of the input data and obtaining the output data, a complete analysis was carried out in the transversal (plate) and axial (rod) nodes in the maximum values, that is, in the peak nodes. Thus, it is possible to see the graphs with the temperature in the central region of the rod and the fuel plate as a function of the burn for each time increment. These graphs are shown in Figures 1 and 2, respectively.

Figure 1: *Temperatures in the center of* UO_2 *rods*



It can be seen that the temperatures in the central region of the uranium dioxide tablets are much higher when compared to those of the uranium-molybdenum plates. This occurred even considering that a maximum power density of 11.70 kW/m was applied to the cylindrical rod and 38.36 kW/m was applied to the fuel plate. The differences in the core temperatures of the fuels analyzed are around 200 °C at the beginning of the burn, increasing to approximately 280 °C after 1500 days of burning and reaching its maximum value of 338 °C at the end of the burning, 2190 days. These temperatures rise for each fuel, because the power density has been increased from the beginning to the end of life.

Figure 2: *Temperatures in the center of UMo plates*



For the same critical points mentioned, the following burns were observed for the rod and the plate, respectively:

- after 1500 days of burning: 37.89 MWd/kgU and 17.81 MWd/kgU; and
- at the end of the burn, 2190 days: 58.08 MWd/kgU and 27.30 MWd/kgU.

In this regard, we can highlight the advantages of the UMo monolithic plate, as it has lower temperatures and requires less fuel for the same power density.

In an analysis encompassing the entire burning period, it can be said that the behavior of the plate is more constant, as the release of gases is less and maintains the temperature with sensitive changes in all increments of time, this is due to the greater thermal conductivity of metallic fuel when compared to that of ceramic material.

Thus, it can be concluded that the temperatures in the rods are higher than in the fuel plate and that they hardly increase in the plate, which leads to lower fission gas releases.

In the theme related to the design bases (limits and design conditions) of the rods, it can be said that they were met, since the temperatures did not exceed the limit of 2500°C for UO_2, reaching a maximum of 772.2 ºC.

For the UMo plates, it can be seen that these requirements were met as well, although in some moments, after 750 days of irradiation, the temperature at the heart of the fuel has exceeded 420 ºC, there was no evidence of excessive release of gas or existence of gap between the fuel and the cladding, as stated in the output data of the PADPLAC code.

In this case, for temperatures exceeding 420 ºC, the release of fission gases in the uranium-molybdenum fuel is considered to be significant and must be strictly considered in the performance analysis of the fuel plate.

Under normal operating conditions, there is a closed contact between fuel and cladding, a situation that must remain for a long period in the operation of the reactor. However, the effects arising from the thermal expansion of the cladding and the fuel, the release of fission gases and the swelling of the fuel, as well as the appearance of bubbles arising from the interaction between the fuel and the cladding, can cause the creation of voids in this interface. In this condition, a significant reduction in heat transfer occurs at the interface between the fuel and the cladding.

Knowing that in the ideal condition of contact, all heat transfer occurs by conduction due to the union between the surfaces, in this partial contact there will be a combined heat transfer process, covering conduction in the contact regions and conduction in the gas in regions where contact does not occur. Thus, in this situation where the heat transfer at the interface between the fuel and the cladding has been impaired, the temperature at the heart of the fuel plate will tend to increase, thereby intensifying the process of releasing fission gases [9].

Due to the appearance of bubbles, which provides mild and/or severe deterioration at the interface between the fuel and the cladding, there is an increase in thermal resistance in this contact, thus causing an increase in temperatures at the heart and edge of uranium-molybdenum. This variation in temperature will cause an increase in the release of fission gases causing the distance between the inner face of the cladding and the outer face of the fuel plate, resulting in the appearance of a gap between the surfaces of both parts. Such behavior is intrinsically linked to the absence of empty spaces in the UMo fuel to accommodate fission gases. Therefore, only a small amount of fission gases released is able to provide an increase in the internal pressure in the compartment, as a result, a high internal pressure appears in the fuel plate, which can be higher than the external pressure of the coolant, a fact that can result in the gap opening.

If this fault condition is reached, the fuel plate can no longer be considered operational, mainly due to two significant factors.

The first one deals with the structural issue of the failure, where it can be said that local instability causes a convex deformation/deflection in the entire thickness of the fuel plate cladding, promoting the narrowing of the cooling channel. Such a reduction leads to the appearance of two dysfunctions:

- loss and / or reduction of cooling conditions for the entire fuel plate that has the fault and also for the adjacent plate; and
- increase in the speed of the coolant liquid in the region where the cooling channel narrows, thus causing the appearance of vibrations in the fuel plates that are covered by this decrease in the channel thickness.

The second factor is the thermal hydraulic issue of the failure, so it can be said that the increase in clearance between the outer face of the fuel and the inner face of the cladding, provides a reduction in heat transfer in the fuel-cladding assembly and this causes rising fuel temperature. This scenario contributes to an increase in the release of fission gases, which in turn will increase the internal pressure of the fuel compartment, thus increasing the deflection already existing in the cladding, with the potential for the appearance of cracks in this critical region, which may cause contamination of the coolant.

3.2. Cladding temperature

A comparison was made with the temperatures on the outer wall of the zircaloy cladding for the two types of fuel used, as shown in Figures 3 and 4.

Analyzing the figures, it can be seen that the temperatures on the outside of the fuel cladding are slightly higher when compared to the fuel plates. Thus, such thermal variations are 16 ºC at the beginning of the burning, increasing to approximately 27 ºC at the end of the firing. In this case, the difference is not so significant with a representation of approximately 5.3% to 8.3% in absolute values of the temperature variation.

Figure 3: *Temperatures in cladding rod*

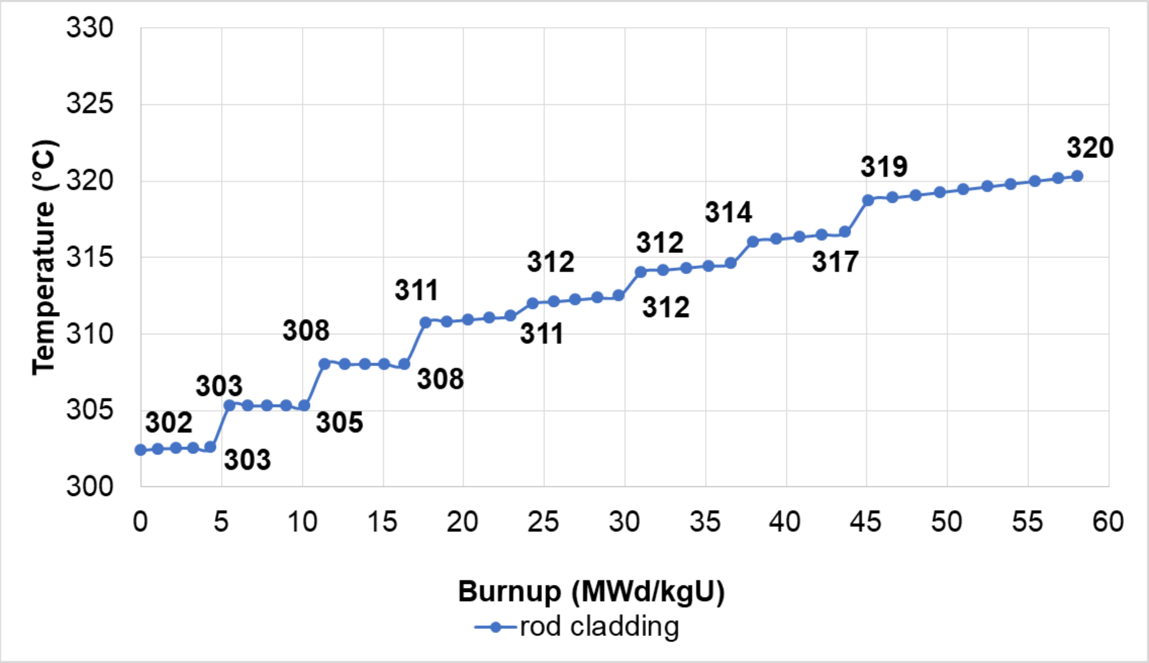
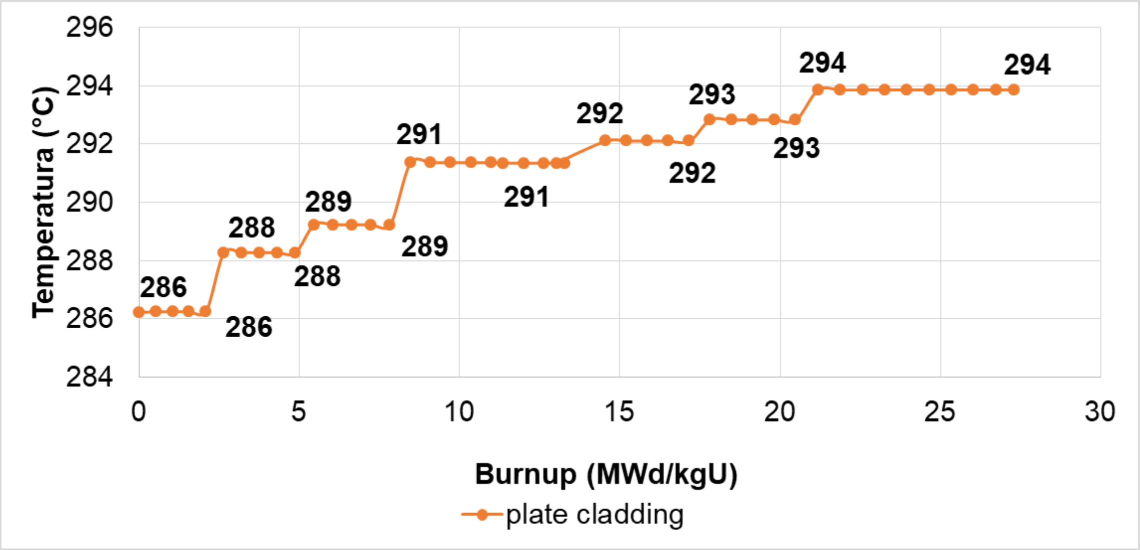


Figure 4: *Temperatures in cladding plate*



In the theme related to the design bases (design limits and conditions), when the deformation of the fuel rod cladding is verified, it can be said that these conditions meet their respective design limits, since at no time have the maximum deformations been reached (deformations less than 1%), according to FRAPCON code output data.

As for the UMo plates, it can be seen that these requirements were met, as for this type of fuel the temperature at the core/cladding of the fuel of 420 ºC is considered as limiting. A maximum of 239.9 ºC was observed for the fuel plate end, although at the core we noticed higher temperatures, as seen previously. Thus, analyzing the PADPLAC output data, it was found that there was no excessive release of gases, thus there was no convex deformation of the cladding plate that could result in the narrowing of the cooling channel or a gap between the fuel surface and the cladding, according to the subject addressed in the previous item.

3.3. Coolant temperature

A comparison was made between the coolant outlet temperatures in the two types of reactor under analysis, fuel rod type and fuel plate type, in order to verify the efficiency of the generation and heat transfer by the UMo plates, since these properties are already enshrined in applications with UO_2 fuel rods. The Figures 5 and 6 show these informations.

It is possible to see that the coolant outlet temperatures are substantially higher in the rod reactor when compared to the plate reactor. These thermal variations are of the order of 9 ºC at the beginning of the irradiation, increasing to approximately 11 ° C after 1050 days of burning and reaching its maximum value of 12 ° C between the time 1750 days until the end of the irradiation.

Based on these data on temperature variations, it can be noted that there were small differences, presented with a minimum variation of 3.2% to a maximum of 4.3%, showing the probable efficiency and effectiveness in the implantation of fuel plates for operation at the rated power analyzed.

Figure 5: *Coolant temperature (rod)*

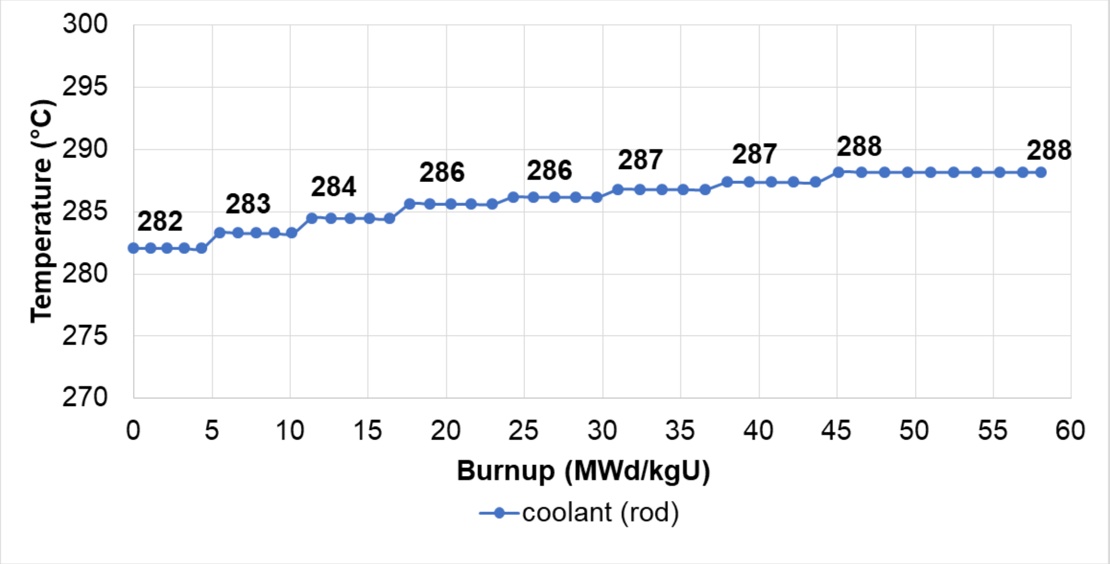
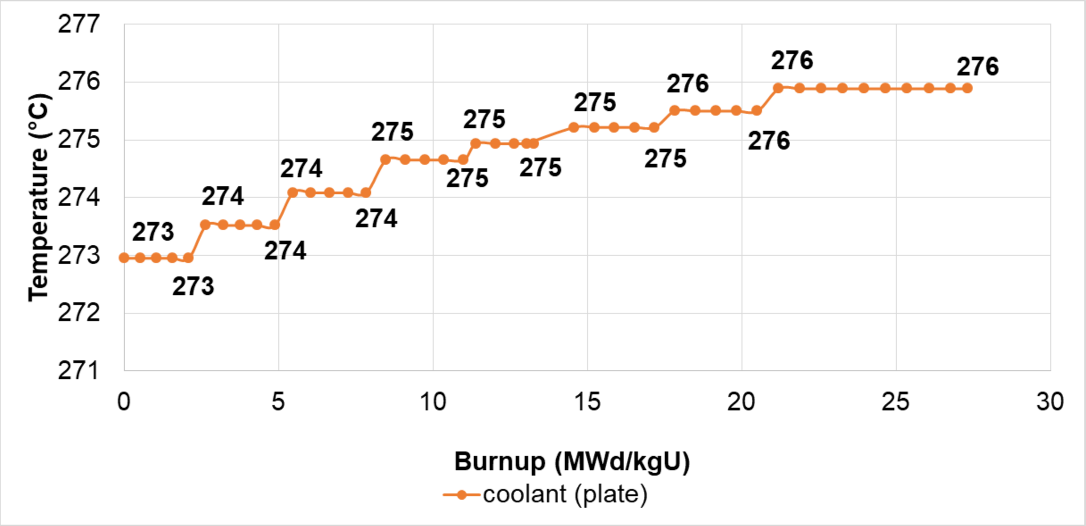


Figure 6: *Coolant temperature (plate)*



4. CONCLUSION

A comparative analysis in terms of the performance under irradiation of cylindrical uranium dioxide fuel rods and monolithic uranium molybdenum fuel plates in reactors to pressurized light water was carried out using the fuel performance codes FRAPCON and PADPLAC-UMo, respectively. In this analysis case was possible to see that the fuel plate in relation to the fuel rod showed a higher power density and behaved better heat transfer to the coolant, leading to a reduction in fuel temperatures. The higher power density allows the production of cores with more compact fuel plates.

In the case of the 58 MW small reactor analyzed, the difference in power density in relation to the fuel plate (38.36 kW/m) and the fuel rod (11.70 kW/m) does not deteriorate the performance of the plant. Even with higher power densities, the plate type fuel had lower temperatures, although it reached its design limit in relation to its operating temperature (420 ºC). No design limits have been reached on the rod fuel. The PADPLAC-UMo code does not perform mechanical calculations, such as those provided by the FRAPCON code, but it can be said that the plant's safety is increased in the case of reactors with plate type fuel, due to the greater capacity to resist dynamic loads when compared to the resistance of the fuel rods.

REFERENCES

[1] TRAVELLI, A. **The RERTR Program**. Argonne National Laboratory. ENS RRFM 97 Transactions. 1997. 8 p.

[2] MEYER, M. K. et al. **Irradiation Behavior of Uranium-Molybdenum Dispersion Fuel: Fuel Performance Data from RERTR-1 and RERTR-2**. Argonne National Laboratory. 1999. International RERTR Meeting. 12 p.

[3] MEYER, M. K. et al. **Metallographic Analysis of Irradiated RERTR-3 Fuel Test Speciments**. Argonne National Laboratory. 2000. Presented in 2000 International RERTR Meeting. 13 p.

[4] MEYER, M. K. et al. **Irradiation Performance of U-Mo Monolithic Fuel**. Nuclear Engineering and Technology, Vol. 46, Nº 2. 2014. 14 p.

[5] WOOLSTENHULME, N. E.; GLAGOLENKO, I.; RABIN, B. H. **Design of Irradiation Tests for Monolithic Fuel Qualification.** RERTR 2015. 36 International Meeting on Reduced Enrichment for Research and Test Reactors. 2015. 10 p.

[6] GEELHOOD, K. J.; LUSCHER, W. G.; BEYER, C. E., FRAPCON 3.4: **A Computer Code for the Calculation of Steady-State Thermal-Mechanical Behaviour of Oxide Fuel Rods for High Burnup.** NUREG/CR-7022, Vol. 1. PNNL-19418, 2011. 134 p.

[7] REIS, Regis. **Desenvolvimento de um programa computacional para análise do desempenho sob Irradiação de placas combustíveis monolíticas de urânio-molibdênio em reatores de potência.** 2019. Tese de Doutorado. IPEN-USP. 2019. 140 p.

[8] ANDRZEJEWSKI, C. S. **Avaliação de Alternativas de Combustível Tipo Placa para Reatores de Pequeno Porte**. Dissertação de Mestrado. IPEN. 2005. 83 p.

[9] WACHS, D. et Al. **Blister Threshold Based Thermal Limits for the UMo Monolithic Fuel System.** Idaho National Laboratory - RERTR 2012. INL/CON-12-27501. 2012. 13 p.

[10] BURKES, D. E. et Al. **The Influence of Cladding on Fission Gas Release from Irradiated U-Mo Monolithic Fuel.** Journal of Nuclear Materials 486. 2017. Pág. 222– 233