Analysis of a Sodium-Cooled Fast Reactor during ULOF and UTOP transients using the AZTHLIM code

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Abstract

The main goal of this work is to perform an analysis of a Sodium-cooled Fast Reactor using the AZTHLIM code, simulating two transient events: an Unprotected Loss-of-Flow (ULOF) and an Unprotected Transient-Over-Power (UTOP). The AZTHLIM code is a thermofluid code whose main objective is to describe the processes of heat transfer between the fuel rod and the coolant, developed as part of the AZTLAN Platform project. The core design used in this work has the characteristics of European Sodium Fast Reactor (ESFR), the nominal power of 3600 MWth and fuel based on mixed oxides of uranium and plutonium. For the transient analysis, it is applicate a reduction of the coolant flow to simulate a ULOF transient, using 50% and 10% of the nominal flow. In the case of UTOP transient, it is considered an increase of 100 pcm and 175 pcm in reactivity. For ULOF transients, the maximum sodium temperature obtained is 964.18 K, when the inlet flow is 10% of nominal flow. And for the UTOP transient, 875.72 K was the maximum temperature with an increase of 175 pcm.

1. INTRODUCTION

The Generation IV International Forum (GIF) identified six technologies of reactors [1], these reactors are a potentially sustainable energy source, particularly in terms of waste management and nuclear fuel optimization [2]. Two of these technologies are nuclear Fast Reactors cooled by liquid metals; Sodium-cooled Fast Reactor (SFR) and Lead-cooled Fast Reactor (LFR). The SFR is the technology of GIF that has around 55 years of technological experience, and many countries have investigated and built it [2]. The Sodium-cooled Fast Reactor is one of the most promising candidates to meet goals of the GIF [3].

To study the phenomena in the nuclear reactors core, different codes have been developed to model these phenomena, such as SIMMER-III [4-6]. In some cases, it has been possible to

compare the results of the codes with experimental data, as shown in the work of Droin, *et al*. [7]. The main goal when developing nuclear reactors is the safety, for this reason, it is necessary to be able to simulate transitory events, such as the ULOF and UTOP, to identify the main challenges [8-10]. In Mexico, some institutions are working to develop a set of codes to describe nuclear reactor phenomena, one of these efforts is the AZTLAN Platform project, where the AZTHLIM code belongs to.

The AZTLAN Platform project [11] is a national initiative led by the National Institute of Nuclear Research which gathers the main public institutions of higher education in Mexico: The Autonomous Metropolitan University, the National Polytechnic Institute, and the National Autonomous University of Mexico and the. It consists in the development of a modeling platform for the analysis and design of nuclear reactors. This project aims to modernize, improve and integrate the neutronic, thermohydraulic/thermofluid and thermomechanical codes, developed in the national institutions, is a platform integrated and developed by Mexican experts. The AZTHLIM code is a thermofluid code whose main objective is to describe the processes of heat transfer between the fuel rod and the coolant for SFR.

In this work the AZTHLIM code is used to analyze a SFR in steady state and in ULOF and UTOP transients, considering the neutron, heat transfer and thermofluid processes. The design considered for these exercises is the ESFR [2] reactor. The steady state shows a temperature profile of the fuel, gap, cladding, and sodium. For the transient analysis, it is applied a reduction of the coolant flow to simulate a ULOF transient, using 50% and 10% of the nominal inlet flow. In the case of UTOP transient, it is considered an increase of 100 pcm and 175 pcm in reactivity.

1.1. Core Specifications

The core design used in this work has the characteristics of European Sodium Fast Reactor (ESFR) core [2]. In Table I, the main parameters of the core and the values of radius and length of the fuel pin are presented. The core design is a fuel based on mixed oxides of uranium and plutonium.

Table I. Core parameters of ESFR [2]

Core parameters			
Thermal power	3600 MWth	Inner central hole radius	0.1257 cm
Fuel type	Pins/Pellets	Fuel slug radius	0.4742 cm
Active height	1.00566 m	Inner radius of cladding	0.4893 cm
Core inlet temperature	668.15 K (395°C)	Outer radius of cladding	0.5419 cm
Core outlet temperature	818.15 K (545 °C)	Pin to pin distance	1.1897 cm
Average fuel temperature	1500.15 K (1227°C)	Pin per assembly	271
Inlet flow	19,000 kg/s	Fuel assemblies	453

1.2. Definition of Transients (ULOF and UTOP)

In general, transients could be protected or unprotected. Protected transients correspond to accidents when the reactivity control system works as designed and the initiating event of the transient is followed by SCRAM. During the unprotected transients, however, there is no SCRAM. The classical unprotected transients under consideration are Unprotected Loss-of-Flow (ULOF), Unprotected Transient-Over-Power (UTOP) and Unprotected Loss-of-Heat-Sink (ULOH) [7, 9, 12].

A ULOF (Unprotected Loss-of-Flow) transient is initiated when the power of the primary pump is lost, or a shaft beak occurs. Due to the lack of SCRAM, the only mechanism for decreasing the reactor power is the sum of reactivity feedbacks [12].

A UTOP (Unprotected Transient-Over-Power) transient is initiated when a control rod begins to move out of the core; the control system does not induce SCRAM and the pumps maintain the nominal coolant flow through the core [12].

2. AZTHLIM CODE DESCRIPTION

The AZTHLIM code is a thermal-fluid code, i.e. it models the heat transfer processes in the coolant (liquid metal) of a fast reactor, considering the heat transfer phenomena in the fuel rod and the neutron processes.

The code is constituted by two main modules: module (1) contains the heat transfer process in the fuel rod, considering the fuel pellet, gap and cladding, and in module (2), the heat transfer in the coolant is modeled from mass, momentum and energy balances. For this work, neutron point kinetics equations were used to describe the neutronic process.

In the following section the mathematical models used, correlations for the properties of the materials, and the solutions methods are described.

2.1. Neutronic Model

The neutron density is calculated with neutron point kinetics equations with six precursors of delayed neutrons:

$$\frac{dn(t)}{dt} = \frac{\rho_t(t) - \beta}{\Lambda} n(t) + \sum_{i=1}^{6} \lambda_i C_i(t) \tag{1}$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda} n-(t) - \lambda_i C_i(t)\ , \qquad \text{for} \qquad i = 1,2,3,...,6 \tag{2}$$

$$\rho_t = \rho_0 + \underbrace{K_D \ln\left(\frac{\langle T_f \rangle_{out}}{\langle T_f \rangle_{in}}\right)}_{\text{Doppler}} + \underbrace{\alpha_f \Delta\langle T_f \rangle}_{\text{Fuel expansion}} + \underbrace{\alpha_c \Delta\langle T_c \rangle}_{\text{Clad expansion}} + \underbrace{\alpha_s \Delta\langle T_s \rangle}_{\text{Sodium expansion}} \tag{3}$$

where ρ_t is the total reactivity, β is the total fraction of the delayed neutron, Λ is the mean neutron generation time, λ_i is the decay constant of delayed neutron precursor, C_i is the concentration of the *i*-th delayed neutron precursor. The subscripts $f, g, c,$ s correspond to fuel, gap, clad and sodium respectively. The nuclear parameters (Table II) used in this study were obtained using the stochastic code Serpent version 2.1.28 [13].

The total fraction of delayed neutrons is given by $\beta = 0.00432039$, and the mean neutron generation time is $\Lambda = 4.48274 \times 10^{-7} s$.

Table II. **Nuclear parameters.**

Group	$\lambda_i (s^{-1})$	β_i
1	0.0127023	8.78147×10^{-5}
2	0.0301099	8.16105×10^{-5}
3	0.112331	6.51854×10^{-4}
4	0.327449	1.7707×10^{-3}
5	1.22596	7.88203×10^{-4}
6	8.14883	2.05715×10^{-4}

The total reactivity is given by Eq. (3), that assumes various contributions: initial reactivity margin (ρ_0), Doppler effect, fuel expansion, cladding expansion, and the effect of sodium density whose parameters are: Doppler constant $K_D = -834.66 pcm$, reactivity coefficients by fuel expansion $\alpha_f = -0.303 pcm K^{-1}$, by clad expansion $\alpha_c = 0.0405 pcm K^{-1}$, and by sodium expansion $\alpha_S = 0.4505 pcm K^{-1}$ [13]. It is important to note that the α_c includes the effects of the axial and radial expansion of the clad.

The changes in the average temperature in Eq. (3) are defined as: $\Delta\langle T_f \rangle = \langle T_f \rangle - \langle T_f \rangle_0$ $\Delta\langle T_c \rangle = \langle T_c \rangle - \langle T_c \rangle_0$, and $\Delta\langle T_S \rangle = \langle T_S \rangle - \langle T_S \rangle_0$ [14], where the subscript 0 represents the reference temperature, and $\langle \bullet \rangle$ represents the average temperature on volume in the fuel core, clad and sodium. The thermal power in the subchannel is given by:

$$P(t, z) = P_0 n(t) \psi(z) \tag{4}$$

where P_0 is the nominal thermal power per fuel rod, $n(t)$ is the neutron density, and $\psi(z)$ is the axial power distribution. In this work, the axial neutron flux profiles within the active height were obtained from the work given by Aufiero et al. [15].

The numerical solution of the neutronic power considers two methods. The first is the Runge-Kutta 4th order method, applied for the numerical solution of neutrons density, given by Eq. (1),

and the second is the Euler method, applied for the numerical solution of a concentration of the *i-th* neutron delayed precursor, given by Eqs. (2 and 4). The step length used in this work was 0.0001s.

2.2. Fuel Heat Transfer Model

The fuel mathematical model calculates the heat transfer in annular fuel pellets, as it is illustrated in Figure 1. In the model the inner hole is not considered. The temperature distribution in the annular fuel pin, initial and boundary conditions are given by:

$$(\rho Cp)_f \frac{\partial T_f}{\partial t} = \frac{k_f}{r}\frac{\partial}{\partial r}\left(r\frac{\partial T_f}{\partial r}\right) + q'''(t,z), \qquad Fuel \qquad r_a \le r \le r_f \qquad (5)$$

$$(\rho Cp)_g \frac{\partial T_g}{\partial t} = \frac{k_g}{r}\frac{\partial}{\partial r}\left(r\frac{\partial T_g}{\partial r}\right), \qquad Gap \qquad r_f \le r \le r_g \qquad (6)$$

$$(\rho Cp)_c \frac{\partial T_c}{\partial t} = \frac{k_c}{r}\frac{\partial}{\partial r}\left(r\frac{\partial T_c}{\partial r}\right), \qquad Clad \qquad r_g \le r \le r_c \qquad (7)$$

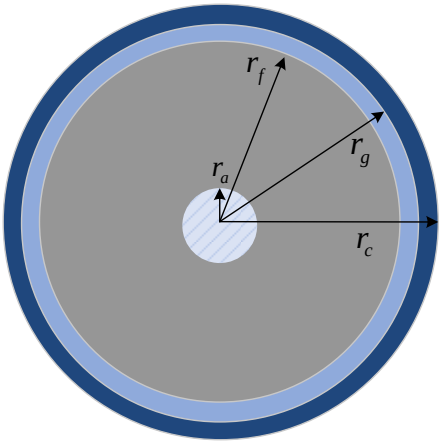

Figure 1. **Annular fuel pellet.**

The initial condition is given by $T(r,0) = f(r)$, and the boundary conditions are:

$$\frac{dT_f}{dr} = 0, \qquad at \qquad r = r_a \qquad (8)$$

$$-k_g \frac{dT_g}{dr} = h_g(T_f - T_g), \qquad at \qquad r = r_f \qquad (9)$$

$$-k_c \frac{dT_c}{dr} = h_g (T_g - T_c), \qquad at \qquad r = r_g \tag{10}$$

$$-k_c \frac{dT_c}{dr} = h_S (T_c - T_S), \qquad at \qquad r = r_c \tag{11}$$

In these equations ρ is the density, Cp is the specific heat, k is the thermal conductivity, h_S is the sodium heat transfer coefficient, h_g is the gap conductance, and $q'''(t,z)$ is the heat source given by:

$$q'''(t,z) = \frac{P(t,z)}{V_f} \tag{12}$$

where, $P(t,z)$ is the subchannel power given by Eq. (4), and V_f is the fuel volume. The physical properties of fuel (considered 0.15 in a molar fraction of plutonium oxide and 0.85 of U-238) as a function of temperature are presented in Table III. Table IV shows the physical properties of the gap (helium) and cladding (SS T91).

Table III. **Fuel properties** [17].

Property	**Correlation**
Density ($kg\,m^{-3}$)	$\rho_f = \dfrac{11043.5}{\left(9.9672\times10^{-1} + 1.179\times10^{-5}T - 2.429\times10^{-9}T^2 - 1.219\times10^{-12}T^3\right)^3}$
Specific heat ($J\,kg^{-1}\,K^{-1}$)	$Cp_f = 0.85\times\left[\dfrac{90.998\times10^6 A}{(A-1)T^2} + 1.6926\times10^{-2}T + \dfrac{1.620\times10^{12}}{T^2}\right] + 0.15\times\left[\dfrac{111.275\times10^6 B}{(B-1)T^2} + 2.9358\times10^{-2}T\right]$
Thermal conductivity ($W\,m^{-1}\,K^{-1}$)	$k_f = 1.158\times\left(\dfrac{1}{0.1205 + 2.6455\times10^{-4}T} + \dfrac{6400\times e^{-16.35/\tau}}{\tau^{5/2}}\right)$ $A = e^{548.68/T};\quad B = e^{-18541.7/T};\quad \tau = T/1000$
Heat transfer coefficient ($W\,m^{-2}\,K^{-1}$)	$h_g = 5000$ [18]

The heat transfer coefficient of Eq. (11) is calculated by $h_S = k_S\,\mathrm{Nu}\,D_h^{-1}$, where k_S is the thermal conductivity (given in Table IV), D_h is the equivalent diameter, and the Nusselt number for each fuel-assembly arrangement, which correlation is given by Todreas and Kazimi, [16]:

$$\mathrm{Nu} = 7 + 0.025\left(\mathrm{Pr}\,\mathrm{Re}\right)^{0.8} \tag{13}$$

where Pr and Re are the Prandtl and Reynolds numbers, respectively.

Table IV. **Gap and clad properties** [18].

Property	Gap	Cladding
Density $(kg\,m^{-3})$	2.425	77000.0
Specific heat $(J\,kg^{-1}K^{-1})$	5191.0	622.0
Thermal conductivity $(W\,m^{-1}K^{-1})$	$15.8\text{x}10^{-4}\,T^{0.7}$	26.0

The annular fuel pellet temperature distribution is obtained considering nineteen radial nodes at each of the twenty-four axial nodes in the core. Ten nodes were considered in the fuel, four nodes in the gap, and five in the clad. The differential equations described previously are transformed into discrete equations using the control volume formulation technique in an implicit form [19].

2.3. Thermofluid Model

The thermofluid in the core is modeled with mass, energy and momentum balance that considers thermal expansion effects whose physical properties are given in Table V.

$$\alpha_{th}\,\rho_S\,\frac{dT_S}{dt} + \frac{\partial G}{\partial z} = 0\,, \qquad \textit{Mass balance} \tag{14}$$

$$\frac{\partial T_S}{\partial t} = \frac{P_m h_S\left(T_c - T_S\right)}{A_f\,\rho_S Cp_S} - \frac{G}{\rho_S}\frac{\partial T_S}{\partial z}\,, \qquad \textit{Energy balance} \tag{15}$$

$$\frac{\partial G}{\partial t} = -\frac{\xi_{fr}}{2}\left(\frac{G^2}{\rho_S L}\right) - \frac{\partial}{\partial z}\left(\frac{G^2}{\rho_S}\right) - \rho_S g\,, \qquad \textit{Momentum balance} \tag{16}$$

In these equations, α_{th} is the thermal expansion coefficient, G is the mass flux, P_m is the wet perimeter (given by $d_{rod}\left(\pi - 4\right) + 4l_p$), and A_f is the flow area (cross-sectional area). In the momentum balance given by Eq. (16), the friction coefficient is calculated with the following relation:

$$\xi_{fr} = \frac{0.210}{\mathrm{Re}^{0.25}}\frac{L}{D_h}\left[1 + \left(\frac{l_p}{d_{rod}} - 1\right)^{0.32}\right] \tag{17}$$

where the rod pitch used is $l_p = 11.897mm$ [2], d_{rod} is the rod diameter, and A_f is the flow area. The hydraulic diameter for each array is given by:

$$D_h = \frac{4}{\pi d_{rod}} \left(\frac{\sqrt{3}}{2} l_p^2 - \frac{\pi d_{rod}^2}{4} \right) \tag{18}$$

The numerical solution applied to balance equations of mass, energy, and momentum was the Euler method.

Table V. **Sodium physical properties** [20].

Property	Value
Density $(kg\,m^{-3})$	845.0
Heat capacity $(J\,kg^{-1}\,K^{-1})$	1269.0
Thermal conductivity $(W\,m^{-1}\,K^{-1})$	68.8
Viscosity ($Pa \cdot s$ **)**	0.00025

The coupling of the physical processes involves a complex dynamic interaction of variables among the nuclear processes, of fuel heat transfer, and thermofluid. The simulation of nuclear processes with the neutron point kinetics approach is coupled with fuel heat transfer through average temperatures of the fuel and cladding, and at the same time it is coupled with thermofluid through the average temperature of the sodium.

The fuel heat transfer calculations requires the knowledge of the nuclear heat source and the thermal properties of sodium. And the calculations of the thermofluid behavior in the core requires the cladding temperature.

3. SIMULATIONS

3.1. Stationary State

The analysis developed in this work begins with an analysis of stationary state, Figure 2 shows a distribution axial-radial temperature of a fuel rod, in this figure it is possible to identify the differences of temperature between fuel, gap and clad. The maximum temperature is 1912.4 K, it is in the yellow zone at $z = 0.61$ m, and the minimum temperature is 675.52 K, it corresponds to the position of the gap at the beginning of the fuel rod (coordinate $-z$).

With AZTHLIM code it is possible to obtain the radial temperature profiles, in Figure 3 it is showed the profiles at $z = 0$ where the bottom part of the active core is located, at $z = 0.61$ m where the core has the maximum temperature, and at $z = 1.0056$ m at the highest part of the active core. The profile on $z = 0.61$ m is the highest in fuel zone, nevertheless, in the cladding zone the profile is highest in another z coordinates different from $z = 0.61$ m. In safety analysis,

the clad temperature is important, in this work the wall temperature at the end of the active core is 828.93 K.

The distribution of temperature on fuel has a profile such as the axial power distribution, where the highest power is in $z = 0.61$, in the case of cladding profile, the maximum temperature is in the end of active zone due to the direction of the fluid and the accumulation of energy in the coolant.

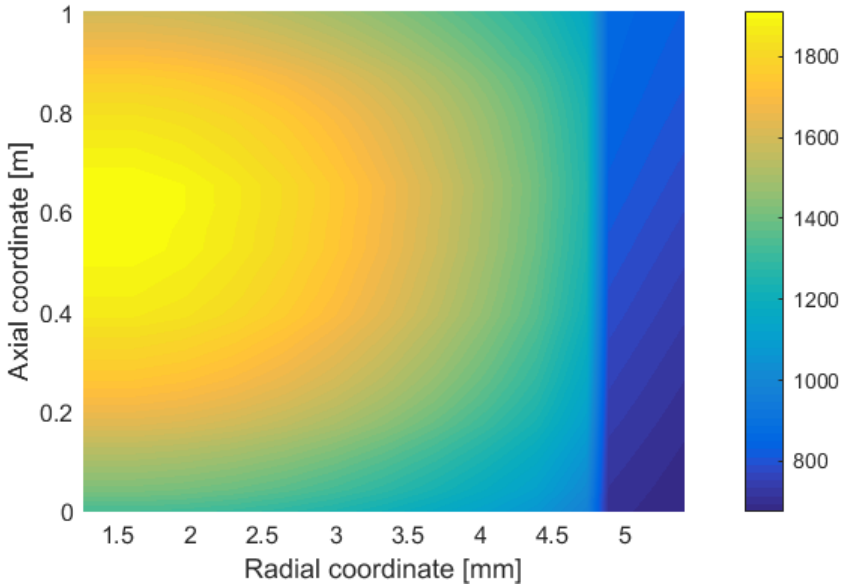

Figure 2. Axial-radial distribution of temperatures in the fuel rod.

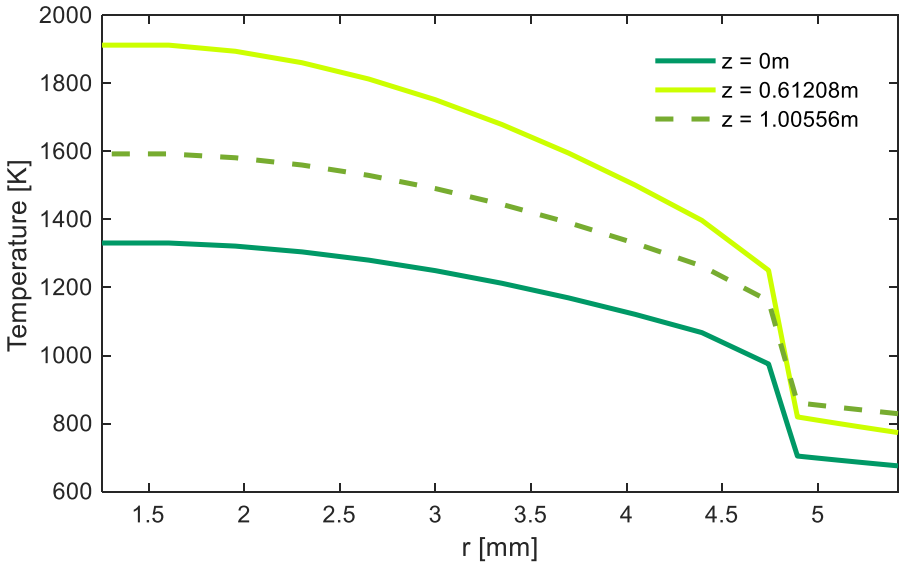

Figure 3. Radial distribution of temperatures in the fuel rod.

The axial profile of temperature is obtained for the fuel, gap clad and sodium, Figure 4 shows the average temperature obtained for each zone. The average temperature of the fuel has the behavior

of the axial profile [15], and the temperature of the clad and coolant has an average difference of 20 K. The maximum average temperature of the fuel (1640.51 K) and gap (1041.30 K) is at $z = 0.61$ m, on the other hand, the maximum average temperature of clad (841.47 K) and sodium (825.77 K) is when $z = 1.0056$ m.

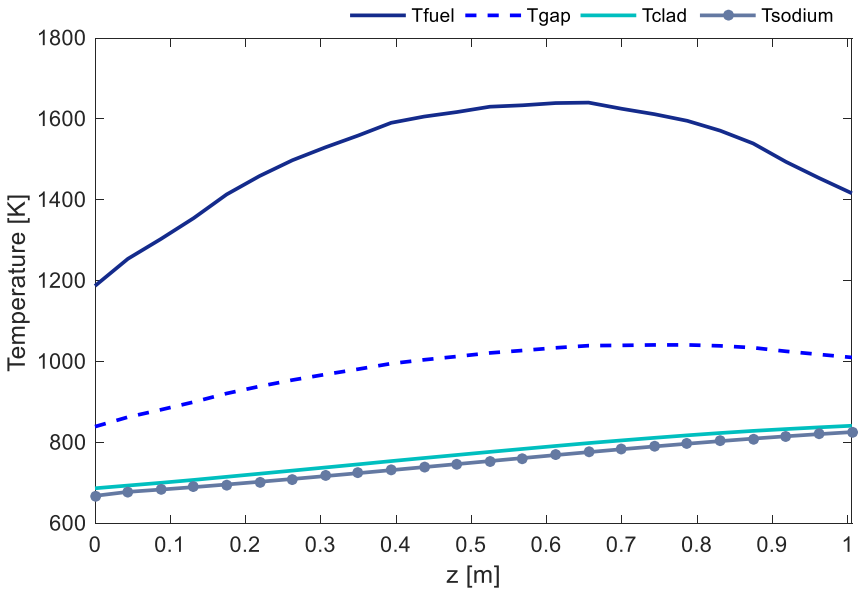

Figure 4. **Axial distribution of temperatures in the core.**

3.2. Analysis of ULOF and UTOP Transients

For the analysis of transients, two events were selected; Unprotected Loss-of-Flow (ULOF) and Unprotected Transient-Over-Power (UTOP). For ULOF transients, in this work the simulation consists in to reduce the flow of coolant until having only 50% and 10% of nominal flow.

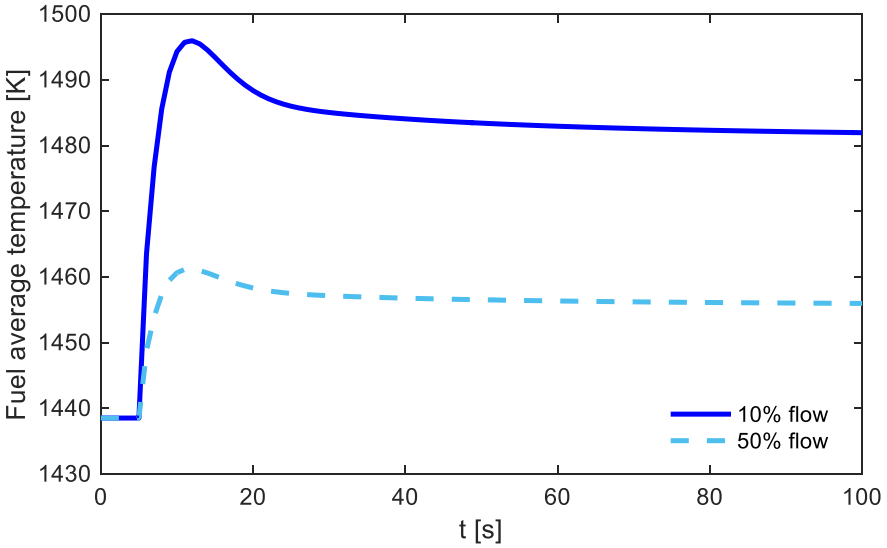

Figure 5. **Fuel average temperature during an ULOF transient.**

Figure 5 shows the fuel average temperature during the transient, in the case for 50% of flow the fuel temperature is 1461.18 K, for 10% of flow is 1495.94 K, having a time to stabilization around 100s.

Is possible to see the effect in the sodium temperature, Figure 6 shows the profiles of temperature at 100s after ULOF transient, the maximum temperature is when the inlet flow is 10% of nominal, 963.98K and for 50% is 881.35 K. In the case of 10% of flow, the temperature is close to sodium boiling temperature.

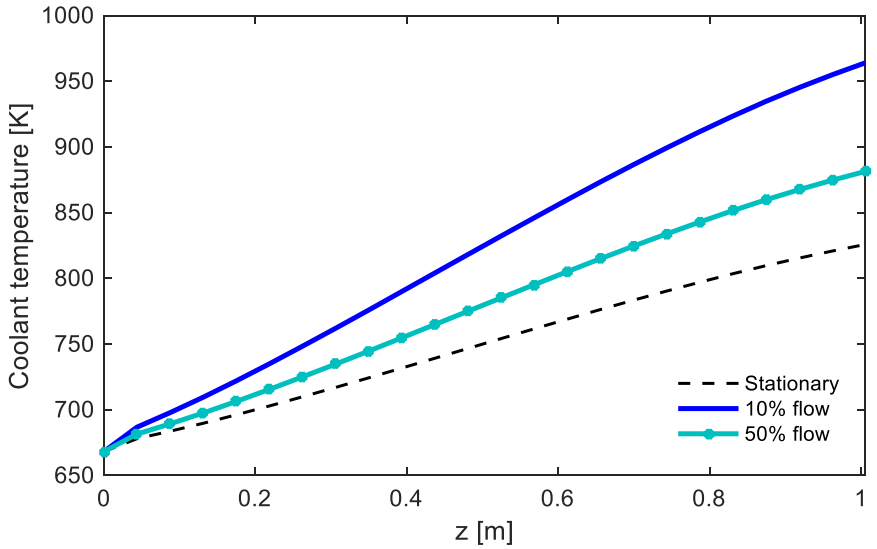

Figure 6. **The behavior of sodium temperature before an ULOF transient.**

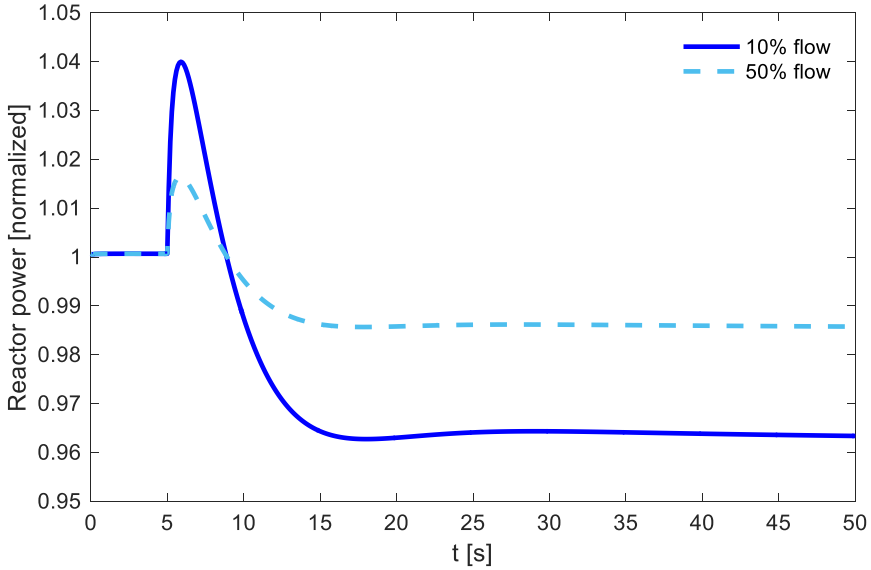

Figure 7. **Reactor power normalized during an ULOF transient.**

In Figure 7 the reactor power is showed, in the beginning of the transient the power increase 4% for 10% of flow and 1.5% for 50%, then, the power decrease to 96% for 10% of flow, and for 50% it decreases 1.2%.

In the case of UTOP transient, a positive reactivity was introduced. Two cases were analyzed: when the reactivity increases 100 pcm and 175 pcm. In this transient, the temperature of the sodium increases as the reactivity increase. Figure 8 shows the fuel temperature during the transient, for the increase of 100 pcm, the fuel average temperature is 1585.51 K. In the case of 175 pcm, is 1698.98 K.

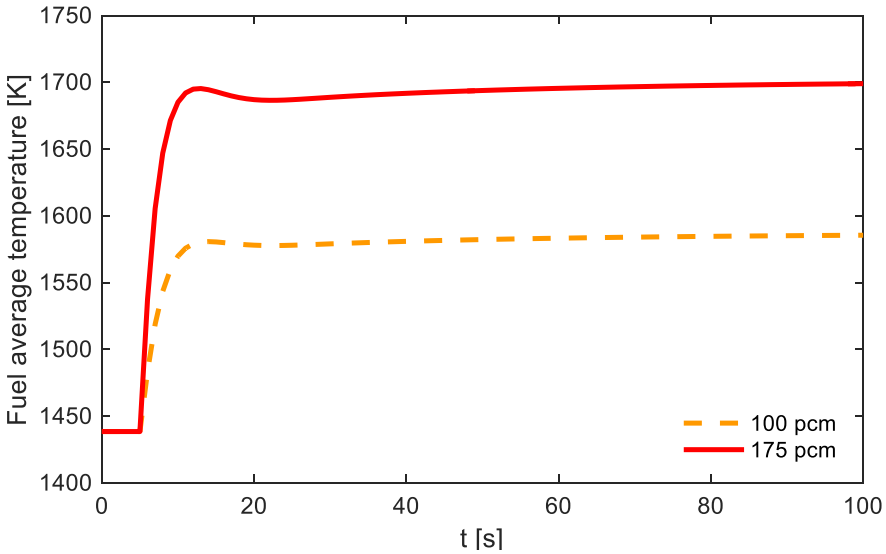

Figure 8. **Fuel average temperature during an UTOP transient.**

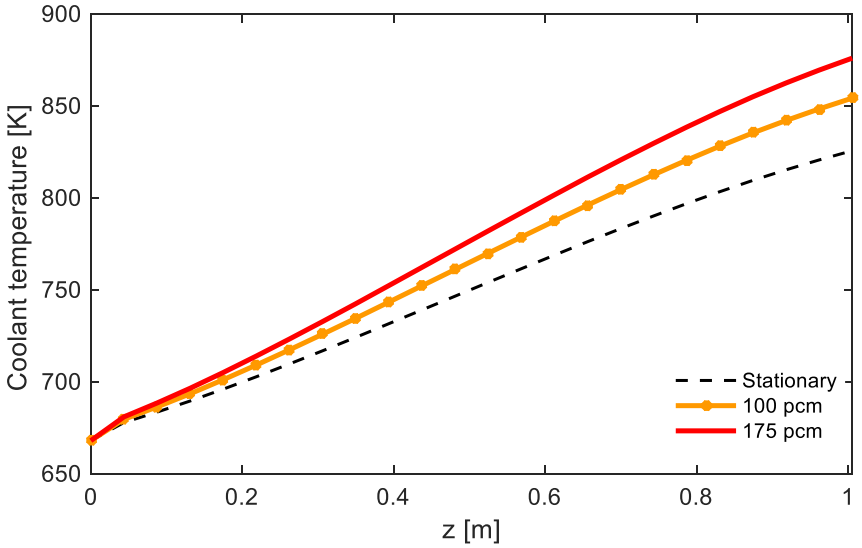

Figure 9. **The behavior of sodium temperature before an UTOP transient.**

Figure 9 shows the temperature of sodium after UTOP. The maximum temperature 876.10 K is with 175 pcm, and with a 100 pcm increase to 854.26 K. These results are similar to the ones reported in the work of Matuzas *et al* [21].

The reactor power is showed in Figure 10, in the beginning of the transient the power increase to 1.70 for 170 pcm and to 1.32 for 100 pcm, then, the power it stabilizes to 1.25 for 175 pcm and in 1.18 for 100 pcm.

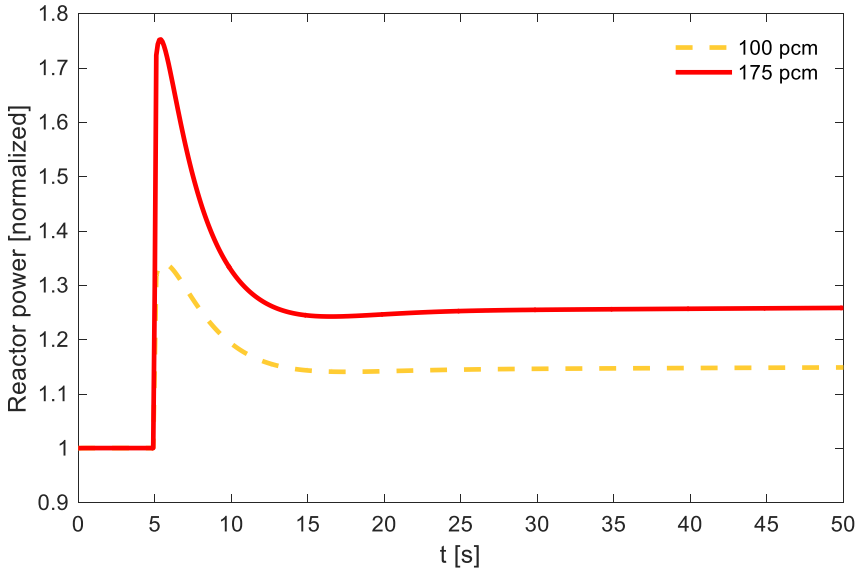

Figure 10. **Reactor power normalized during an UTOP transient.**

4. CONCLUSIONS

The main goal of this work was to analyze a Sodium Fast Reactor using the AZTHLIM code. The analysis was done in stationary state and during ULOF and UTOP transients.

The results of the exercise showed that it is possible to obtain a steady state for a SFR, where it was found that the average temperature of fuel 1506 K, is close to nominal temperature 1500.15 K, reported in Table 1. The calculated outlet coolant temperature was 825.77 K and the nominal was 818.15 K.

In the case of ULOF transients, the maximum coolant temperature obtained was 964.18 K when the inlet flow is 10% of nominal flow. For the UTOP transient, 875.72 K was the maximum coolant temperature with an increase of 175 pcm.

These results are important in the verification of the AZTHLIM code, and gives confidence in it. One of the future activities in the code development is to consider the void effect in the equation of reactivity and in the analysis of thermofluid.

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REFERENCES

1. Generation IV International Forum. "A Technology Roadmap for Generation IV Nuclear Energy Systems". *Issued by the U.S. DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum*, GIF-002-00 (2002).
2. Facchini, A., Giusti, V., Ciolini, R., Tucek, K., Thomas, D., and D'Agata E., "Delailed neutronic study of the power evolution for the European Sodium Fast Reactor during a positive insertion of reactvity". *Nucler Engineering and Design*. **313**, pp. 1-9 (2017).
3. Sun, K.., Krepel, J., Mikityuk, K, Chawla, R., "A neutronics study for improving the sadety and performance parameters of a 3600 MWth Sodium-cooled Fast Reactor", *Annals of Nuclear Energy*, **53**, pp. 464-475 (2013).
4. Kondo, S., Morita, K., Tobita, Y., Kamiyama, K., Brear, D.J., Fischer, E.A., "SIMMER-III: a computer program for LMFR core disruptive accident analysis". *Oarai Engineering Center, Power Reactor and Nuclear Fuel Development, Corporation*. 1996
5. Rimpault, G. et al., "The ERANOS code and data system for fast reactor neutronic analysis". *PHYSOR 2002*, Seoul, South Korea. October 7–10, 2002
6. Qvist, S. and Greenspan, E. "An Autonomous Reactivity Control system for improved fast reactor safety", *Progress in Nuclear Energy*, **77**, pp. 32-47 (2014).
7. Droin, J-B., Marie, N., Bachrata, A., Bertrand, F., Merle, E., and Seiler J.M., "Physical tool for Unprotected Loss Of Flow transient simulations in a Sodium Fast Reactor", *Annals of Nuclear Energy*, **106**, pp. 195-210 (2017).
8. Kruessmann, R., Ponomarev, A., Pfrang, W., Struwe, D., Champigny, J., Carluec, B., Schmitt, D.,and Verwaerde, D. "Assessment of SFR reactor safety issues: Part II: Analysis results of ULOF transients imposed on a variety of different innovative core designs with SAS-SFR", *Nuclear Engineering and Design*, **285**, pp. 263-283 (2015).
9. Tesinsky, M., Zhang, Y., and Wallenius, J. "The impact of americium on the ULOF and UTOP transients of the European Load-cooed SYstem (ELSY)", *Annals of Nuclear Energy*, **47**, pp., 104-109 (2012).
10. Bertrand, F., Marie, N., Prulhiére, G., Lecerf, J., and Seiler, J.M., "Comparison of the behavoiur of two core designs for ASTRID in case of severe accidents", *Nuclear Engineering and Design*, **297**, pp. 327-342 (2016).
11. Gómez-Torres, A. et al., "Mexican platform for analysis and design of nuclear reactors", *Proc. ICAPP 2015*, Nice, France (2015).
12. Shahzad, M., Qin, L., and Imam S., "Analysis of lead-cooled fast reactor using a core simulator", *Progress in Nuclear Energy*, **104**, p. 229-241 (2012).
13. Leppänen, J., et al. "The Serpent Monte Carlo code: Status, development and applications in 2013." *Ann. Nucl. Energy*, **82**, pp 142-150 (2015).
14. Waltar, A.E., Todd, D.R., Tsvetkov, P.V., *Fast Spectrum Reactors*, Springer, New York, 720p. (2012.)

15. Aufiero, M., Cammi, A., Fiorina, C., Luzzi, L., Sartori, A., "A multi-physics time-dependent model for the Lead Fast Reactor single-channel analysis ", *Nuclear Engineering and Design*, **256**, pp. 14-27 (2013.).
16. Todreas, N.E., Kazimi, M.S., *Nuclear Systems I: Thermal Hydraulic Fundamentals*. Hemisphere Publishing Corporation, USA, 705 p. (1990).
17. Carbajo, J., Yoder, G., Popov, S., Ivanov, V, "A review of the thermophysical properties of MOX and UO2 fuels". *Journal of Nuclear Materials*, **299**, pp. 181-198 (2001).
18. Glasstone, S. and Sesonske, A., *Nuclear reactor engineering*. Van Nostrand Reinold Company, N.Y. 1981: 805 p.
19. Espinosa-Paredes, G. and Espinosa-Martínez, E.-G., "Fuel rod model based on Non-Fourier heat conduction equation". *Annals of Nuclear Energy* **36**, 680-693 (2009).
20. IAEA "Comparative assessment of thermophysical and thermohydraulic characteristics of lead, lead-bismuth and sodium coolants for fast reactors", *IAEA-TECDOC-1289*. IAEA, VIENNA, 2002.
21. Matuzas, V., Ammirabile, L., Cloarec, L., Lemasson, D., Perez-Martin, S. and Ponomarev, "A. Extension of ASTEC-Na capabilities for simulating reactivity effects in Sodium Cooled Fast Reactor" *Ann. Nucl. Energy*. (2017)