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Sub-channel thermal-hydraulic analysis for VVER-1000 generation III PWR

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Sub-channel thermal-hydraulic analysis for VVER- 1000 generation III PWR

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Abstract. The sub-channel thermal-hydraulic analysis of a nuclear reactor is essential for assessing of its safety aspects. In this paper, the VVER-1000 has been selected as an example of the third generation reactors since it meets most of the international safety standards and because it has been taken as a base for designing the VVER-1200 which is belonging to the III+ generation. A steady state mathematical model has been proposed and solved to validate and assure that the hottest channel temperature limits are satisfied. The various temperature distributions, the critical heat flux and the departure from nucleate boiling ratio (DNBR) for the hottest channel were evaluated. Also, a transient state model has also been presented and solved using the finite difference method with the aid of MATLAB algorithm. An exponential loss of flow rate of the reactor core coolant was triggered from the steady state conditions. We assumed that the neutron flux and the generated power were unchanged during the postulated event. The average core coolant flow time constant was treated as a single parameter expressing the rapidity of the event. A value of 250 seconds time constant was assumed for slow transient, whereas 10 seconds was assumed for fast one. The reactor core was assumed to be protected through the reactor control system and mitigated according to the regular emergency operating procedures. The time dependent temperature distributions were calculated for the cladding of the hottest coolant channel. For each value of the temperature, the response time required for reaching unsafe conditions was evaluated, discussed and presented.
Keywords: Thermal-hydraulics, steady-state, transient state, hottest channel, MATLAB, DNBR, loss of flow, time constant, postulated event.

1. Introduction

The objective of thermal hydraulic analysis is to ensure the operational temperature in the core does not exceed the design limit of temperature. If it can be ensured the hottest channel exhibits the temperature below the core design limit, the remaining channels then will presumably fall within this limit. Hence, particular attention in this research has been paid on the hottest channel to ensure its temperature remains below the design limit. During the continuous operation of the reactor at high power levels, the nuclear heat generated can be permitted to flow through the following auxiliary systems:

a. During the reactor operation, the forced circulation coolant system pumps pool water down through the fuel elements to remove the fission heat from the reactor.
b. A water-to-water heat exchanger transfers the generated heat to a secondary water coolant system, and the primary water is returned to the reactor core.
c. The secondary water carries heated water to the cooling tower which dissipates the heat to the atmosphere. Water from the cooling tower is recirculating through the secondary system.

To ensure reactor safety, the reactor must have sufficient margins during normal operation as well as in all possible accidental events, such as reactivity-induced accidents (RIAs), loss-of-coolant accidents

(LOCAs), loss-of-flow accidents (LOFAs), etc. This paper focuses on LOFA analysis of selected example of VVER-1000 PWR reactor with the aim of investigating the impact of an unprotected LOFA on the reactor safety. A LOFA occurs in a reactor due to many causes, such as loss of off-site power, pump failure, heat exchanger blockage, pipe blockage, etc. The danger of a LOFA is that it could lessen fuel integrity due to overheating that arises from a low heat transfer coefficient in the reactor core. As a LOFA is classified by regulatory bodies as a design-basis accident, necessary precautions need to be taken during the design stage to ensure that the primary coolant system has adequate safety margins against the onset of flow instability and also against occurrence of the nucleate boiling. This paper provides the result of a sub-channel thermal-hydraulic analysis of the typical VVER-1000 reactor core at steady-state and transient conditions using an analysis approach that required predicting the heat produced from the burn-up of the fuel as it transferred through the fuel and cladding materials reaching to the coolant that passes beside the outer surface of the cladding.
The heat transfer coefficient of the light water which is used as a coolant is calculated. Also the equivalent diameter of the coolant channel, Reynolds number, Prandlt number, Nusselt number and coolant temperature are calculated at the average coolant temperature and normal operating coolant pressure. A code which was designed to match with the MATLAB environment has been used to solve both the steady-state and the transient models.

2. Reactor Description and Analytical Approach

2.1. Reactor Description

VVER-1000 reactor is a Russian-type pressurized water reactor. The major difference between the VVER and a Western PWR, in the present study, is the fuel assembly design and the core geometry as the fuel rod of the VVER type includes a central hole as shown in figure1. The specifications of reactor which is selected in this paper are presented in table.1[1]. Specifically, the reactor under study is made up of 163 hexagonal fuel assemblies of three different enrichment, (1.6%, 2.4% & 3.6). In the core analysis, the individual sub-channels are lumped together to give an equivalent flow area and the fuel rods are modeled using a single rod to represent the average behavior of all rods in each channel. In this analysis, a thermal-hydraulic grid of 280 control volume, each assembly-size of 23.6 cm hexagonal horizontal cross-section and axial height of 353 cm was considered.

Thermal-hydraulic calculations for; temperature, enthalpy and critical heat flux (CHF) and also, the Minimum Departure from Nucleate Boiling Ratio (MDNBR) of the hottest channel in the reactor core were calculated. In this case, the Dittus-Boelter correlation is used for heat transfer coefficient calculation in both sub-cooled and saturated nucleate boiling region. In sub-cooled condition, a partition of the total heat flux, from the heated fuel rods, induces the liquid and therefore produces vapor. The latter is formed by adding single-phase convection from the wall to the liquid film, to the nucleate boiling term while the former is remaining convective heat flux from the liquid film to the coolant bulk. In saturated conditions, however, the total heat flux (including direct heating of the coolant) is converted into vapor, so holding the liquid enthalpy at the saturation value [2].

2.2 Steady-State Model

A mathematical model that describes the heat transfer process from the core to the coolant taking into account the geometric specifications of the hexagonal fuel assemblies and its effect on the fluid dynamic calculations of the coolant is discussed through the following analytical equations. The nodalization of the hottest channel that is chosen to execute the calculations is shown in figure 2. In order to investigate the axial temperature distribution of fuel, clad and coolant along the hottest channel, the core power density firstly calculated using the following formula:

$$\left(q'''_{max}\right)_{actual} = \frac{2*1.16\ \mathrm{P\ E_d}}{3\ \mathrm{H\ a^2\ n\ E_R}} \qquad (1)$$

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Where: q'''_{max} .. is the actual maximum heat generated per unit volume of the hottest channel, E_d… is the energy per fission deposited in fuel elements, E_R… is the energy recovered per fission, n… is the no. of fuel rods, a…. Is the fuel radius, H… is the extrapolated height of the fuel rod; P… is the reactor power.

Table 1. Reactor specifications

Reactor core operating conditions	**Value**
Reactor Pressure (MPa)	15.7
Reactor thermal power (MWt)	3000
Inlet coolant flow rate (kg/s)	18600
Coolant inlet temperature (ºc)	289.7
Coolant outlet temperature (ºc)	320
Fuel assembly	
Fuel assembly form	Hexagonal
Number of fuel assemblies in the core	163
No. of fuel rods in the fuel assembly	312
No. of other tubes	19
Fuel rod	
Central hole of fuel pellet diameter (mm)	2.3
Fuel pellet outside diameter (mm)	7.53
Cladding outside diameter (mm)	9.1
Fuel pellet material	UO2
Cladding material	Alloy-110
Fuel rods pitch (mm)	12.75
Maximal temperature of fuel rod cladding external surface (ºc)	350

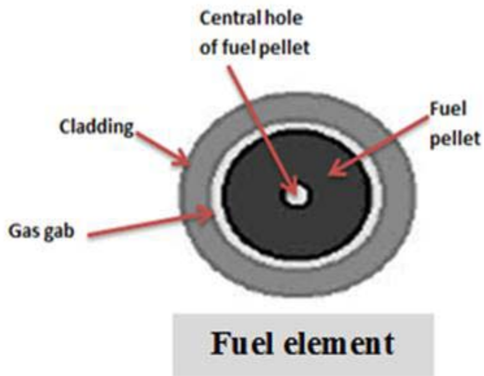

Figure 1. plan view of the fuel rod.

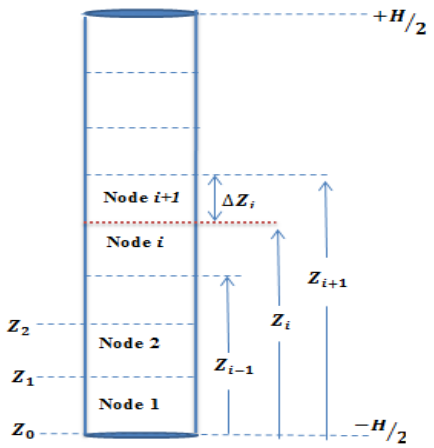

Figure 2. Nodalization of the hottest channel

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Calculation of coolant temperature along the fuel rod of the hottest channel is determined as follows:

$$T_w(z) = T_{w_0} + \frac{q'''_{max} V_f}{\pi \omega C_p}\left[1 + \sin\frac{\pi z}{H}\right] \tag{2}$$

Where;

T_{w_0} .. Is the inlet coolant temperature, ºC,
V_f .. Is the volume of the fueled portion of the rod, m^3,
z .. Is the height of the fuel rod, m,
ω .. Is the rate of flow of the coolant through the coolant channel, kg.s^{-1},
C_p .. Is the specific heat of the coolant, J.kg^{-1}. ºC^{-1}.

And In order to calculate the cladding temperature, the heat transfer coefficient is required to be determined through the Dittus - Boelter correlation as follows [3] :

$$h = C\left(\frac{K}{D_e}\right) R_e^{\,0.8}\, P_r^{\,0.333} \tag{3}$$

$$R_e = \frac{\rho\, V\, D_e}{\mu} \tag{4}$$

$$P_r = \frac{C_p\, \mu}{K} \tag{5}$$

where;
C .. is a constant that can be given by C= 0.026 $(\frac{P}{D})$ - 0.024 for triangular lattices with $1.1 \leq \frac{P}{D} \leq 1.5$. The quantities P and D are, respectively, the lattice pitch and rod diameter,
μ .. is the water viscosity kg.s^{-1} m^{-1},
K .. is the water thermal conductivity,W.m^{-1} ºC^{-1}.
C_p .. is the specific heat of the coolant, J.kg^{-1}. ºC^{-1}.
D_e .. is the equivalent diameter of the coolant channel, m.

Hence, the cladding outside temperature distribution can be calculated from the following equations[4]:

$$T_{co}(Z) - T_w(z) = \frac{q'_{max}}{2\pi R_{co} h} \tag{6}$$

where; R_{co} .. is the clad outside radius, m.

Conduction heat transfer through the cladding material and gap is calculated using Equations (7) and (8), respectively:

$$T_{ci}(z) - T_{co}(z) = \frac{q'_{max}}{2\pi K_c}\ln\left(\frac{R_{co}}{R_{ci}}\right) \tag{7}$$

$$T_{fo}(z) - T_{ci}(z) = \frac{q'_{max}}{2\pi R_g h_{gap}} \tag{8}$$

Where;
$T_{ci}(z)$.. is the clad inner surface temperature at any height (z).
$T_{co}(z)$.. is the clad outer surface temperature at any height (z).
$T_{fo}(z)$.. is the fuel outer surface temperature at any height (z).
R_{ci} .. is the clad inside radius, m.
K_c.. is the clad thermal conductivity,W.m^{-1} ºC^{-1}.

R_g .. is the mean radius in the gap, m.
h_{gap} .. is the effective gap conductance, W.m^{-2} ^{o}C^{-1}.

Moreover, conduction heat transfer in the annular fuel is found by Equation 9:

$$\int_{T_{fo}}^{T_{fi}} K_f(t)dt = \frac{q'_{max}}{4\pi}\left[1 - \frac{\ln\left(\frac{R_{fo}}{R_v}\right)^2}{\left(\frac{R_{fo}}{R_v}\right)^2 - 1}\right] \tag{9}$$

Where; $T_{fi}(z)$.. is the fuel inner surface temperature at any height z.

As the determination of both the maximum fuel and cladding temperatures along the fuel rod of the hottest channel is of great importance in the thermal analysis of the reactor core and for the thermal design specifications, the following equations are utilized to investigate these temperatures [5]:

$$T_{w,max} = T_{w_0} + \frac{2q'''_{max}V_f}{\pi\omega C_p} \tag{10}$$

$$T_{c,max} = T_{w_0} + q'''_{max}V_f\frac{1}{h*A}\left[\frac{1+\sqrt{1+\propto^2}}{\propto}\right], \text{Where } \propto = \pi\ \omega\ C_p\frac{1}{h*A} \tag{11}$$

$$T_{f,max} = T_{w_0} + q'''_{max}V_f\mathrm{R_{th}}_f\left[\frac{1+\sqrt{1+\beta^2}}{\beta}\right] \tag{12}$$

$$\text{As } \beta = \pi\ \omega\ C_p\ \mathrm{R_{th}}_f \text{ and } \mathrm{R_{th}}_f = \frac{1}{4\pi H_e K_f} + \frac{\ln(1+{}^{b}/_{a})}{2\pi H_e K_c} + \frac{1}{h*2\pi(a+b)*H_e} \tag{13}$$

Where;
V_f .. is the volume of the fuel portion, m^3.
A .. is the is the area of the fuel rod, m^2.
a .. is the fuel thickness, m.
b .. is the clad thickness, m.
K_f .. is the fuel thermal conductivity,W.m^{-1} ^{o}C^{-1}.

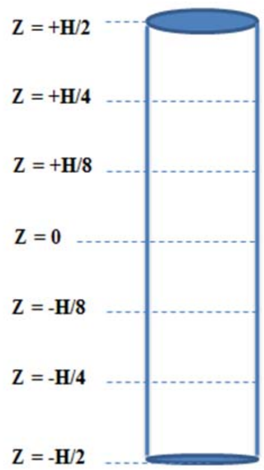

Figure 3. Fuel rod segments

The position of these temperatures along the fuel rod can be determined through the segments of the fuel rod as shown in fig.3 and can be calculated as follows:

$$z_{c,max}=\frac{H_e}{\pi}\tan^{-1}\left(\frac{H_e*h*C}{\pi\omega C_p}\right), \text{Where } C = 2\,\pi\,(a+b) \tag{14}$$

$$z_{f,max} = \frac{\boldsymbol{H_e}}{\boldsymbol{\pi}}\,cot^{-\mathbf{1}}\left(\pi\,\omega\,C_p\,\mathrm{R}_{\mathrm{th}_f}\right) \tag{15}$$

Calculation of DNBR is can be investigated by the determination of the critical and actual heat flux using the most widely used correlation of Jens and Lottes [5]as follows:

$$\text{DNBR}=\frac{q``_c}{q"_{actual}} \tag{16}$$

$$q``_c = C\ \times 10^6\ (\frac{G}{10^6})^m\ \Delta\ T_{sub}{}^{0.22} \tag{17}$$

And C = 0.445 and m = 0.5 for this case. And the T_{sub} can be defined as the difference between the saturation temperature of the coolant and the average coolant temperature along the fuel rod of the hottest channel,

$$q``_{actual} = \frac{q'''_{max}V_f}{C_c} \tag{18}$$

Where; Cc is the circumference of a heated rod.

2.3 Transient-State Model

The transient-state model that describes the thermal-hydraulic behavior of the hottest channel through the loss of flow anticipated transient without scram (ATWS) scenario assumed is described below. The transient-state model is divided into three heat balance equations as given in reference [6]. It simulates the effect of variations of the coolant flow rate on the transfer of the heat generated in the fuel pellets to the coolant through the cladding material. While the power density distribution in hottest channel is assumed to be constant as found under the steady-state operating conditions.

2.3.1 Time dependent coolant temperature distribution.

The time dependent coolant temperature $T_w(z,t)$in the hottest channel can be written as follows:

$$A_w\rho_w C_w\frac{\partial T_w}{\partial t} = 2\pi r_c h_{cw}(T_c - T_b) - \dot{m}C_w\frac{\partial T_w}{\partial z} \tag{19}$$

Where:

T_b .. coolant average temperature within the segment dz.ºC,
A .. cross-sectional area, m^2,
ρ .. density, kg.m^{-3},
C .. specific heat, J.kg^{-1}.ºC^{-1} ,
r_c .. the outer clade radius, m,
h_{cw} .. cladding-coolant heat transfer coefficient, W.m^{-2}. ºC^{-1}.

2.3.2 Time dependent cladding temperature distribution

The time dependent cladding temperature $T_c(z,t)$in the hottest channel can be written as follows:

$$A_c\rho_c C_c\frac{\partial T_c}{\partial t} = 2\pi r_f h_{fc}\left(T_f - T_c\right) - 2\pi r_c h_{cw}(T_c - T_b) \tag{20}$$

Where; h_{fc} is Fuel-cladding heat transfer coefficient = 4 $^{K_f}/_{r_f}$, W.m^{-2}. ºC^{-1} [6].

2.3.3 Time dependent fuel temperature distribution.
The time dependent cladding temperature $T_f(z,t)$in the hottest channel can be written as follows:

$$A_f \rho_f C_f \frac{\partial T_f}{\partial t} = P(z) - 2\pi r_f h_{fc}\left(T_f - T_c\right) \tag{21}$$

Where; $P(z)$ is the linear thermal power rate (W.m^{-1}) and equals Q (z) $= A_f$ $q'''_{\max} \int_{z_{i-1}}^{z_i} \cos\frac{\pi z}{H}\, dz$.

3. Results and discussion

3.1 Steady - state
By solving the previous equations analytically, coolant, cladding and fuel temperatures at different axial heights of the fuel rod (fuel rod is divided to seven segments) can be obtained as shown in Table 2.

Table.2: coolant, cladding and fuel temperatures at different axial heights of the fuel rod.

z	T_{fi}, °C	T_{fo}, °C	T_{ci} , °C	T_{co} , °C	T_w, °C
+H/2	363.1	337.1	331.7	330.5	330
+H/4	1274.1	530.3	375.8	340.3	323.8
+H/8	1547	584.6	384.7	338.7	317.4
Zero	1637.2	598.2	382.4	332.8	309.8
-H/8	1531.8	569.4	369.5	323.5	302.2
-H/4	1246.1	502.3	347.8	312.3	295.8
-H/2	322.9	296.9	291.5	290.3	289.7

And from equation (10) the maximum coolant temperature along the fuel rod can be determined to be equal to (330 ºc) and it could be at the upper end of the fuel rod exactly at (3.53m) from the bottom of the fuel rod. And so from both equations (11) and (14) the maximum cladding outer surface temperature obtained along the fuel rod is equal to (340.3 ºc) and can positioned at about (83 cm) from the center of the fuel rod (about 260 cm from the bottom of the rod).And from equations (12), (13) and (15) the maximum fuel inner surface temperature along the rod could be equal (1637.2 ºc) and could be at 1.6 cm from the center of the rod (about 178 cm from the bottom of the rod) [7] as shown in table 3.

The model of the previous equations is simulated using MATLAB software in order to verify the results found in the literature that shows a good agreement. And this also done for the results of the maximum coolant, maximum cladding and maximum fuel temperatures and their positions along the fuel rod for the hottest channel that is chosen for the reason of that if the thermal design specifications is determined upon which the reactor integrity during steady state operation conditions could be guaranteed [8]. The following analysis of the results obtained from MATLAB simulation would show more explanation for the importance of the thermal analysis of the reactor core with respect to the safety point of view and would also show how near the results are from that found in other references [2].

Table 3. Maximum coolant, cladding and fuel temperatures and their heights from the bottom of the fuel rod.

T	$T, ^{\circ}C$	H, cm
$T_{b,max}$	330	353
$T_{c,max}$	340.3	258
$T_{m,max}$	1637.2	178.3

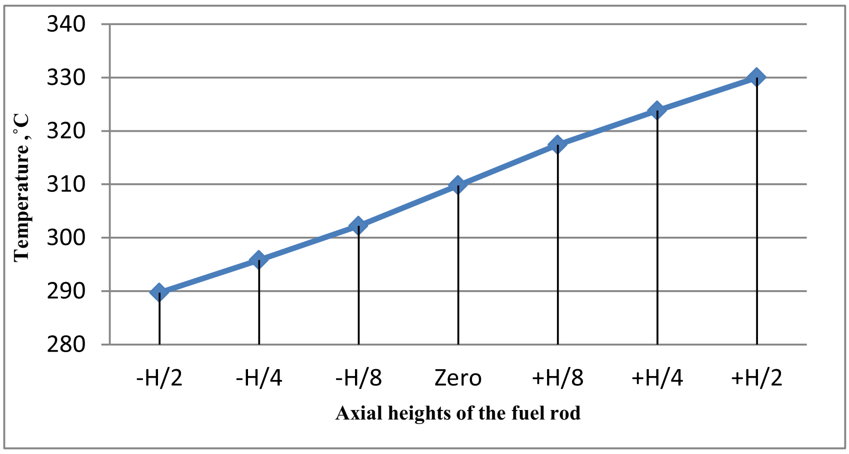

Figure 4. Coolant axial temperature axial distribution

As shown in figure 4, the coolant temperature goes increasing while moving from the bottom side to the upper side of the fuel rod transferring all the heat released from the nuclear fission inside the fuel rod to the coolant `s secondary circuit [9].

It can be noticed from the figures (5) and (6) that both cladding and fuel temperatures rises along the channel and reach maximum values in the upper part of the channel beyond the midpoint of the fuel. There are two reasons that the maximum temperature of the fuel occurs there, rather than at mid channel, where q''' is the greatest.

First, the temperature of the coolant continues to increase past the midpoint. Second, the heat flux q'' is determined only by the value of q''', and this, being a cosine function, decreases very slowly towards z = o. But q'', in turn, specifies the temperature difference ($T_f - T_b$). Therefore, with $T_b$increasing, T_f must also increase to provide the actual value of q'' Further along the channel, q'' begins to drop more rapidly, and T_f eventually decreases. It is this combined effect of a rising T_b and a decreasing q'' that gives rise to and determines the position of the maximum fuel (and cladding) temperature [5].

By Solving equations (16), (17) and (18) the value of the DNBR that is found to be equal about 3.7 can be obtained to be found more greater than the limiting ratio of 1.73 for the case of VVER-1000 reactors and that can be acceptable indication for the thermal design of the core and the thermal-hydraulic analysis`s results of the reactor in steady-state conditions to be so far away from problems that could occur due to the formation of the nucleate boiling on the outer surface of the fuel cladding as a result of the non-homogenous distribution of the temperatures along the fuel rods of the core.

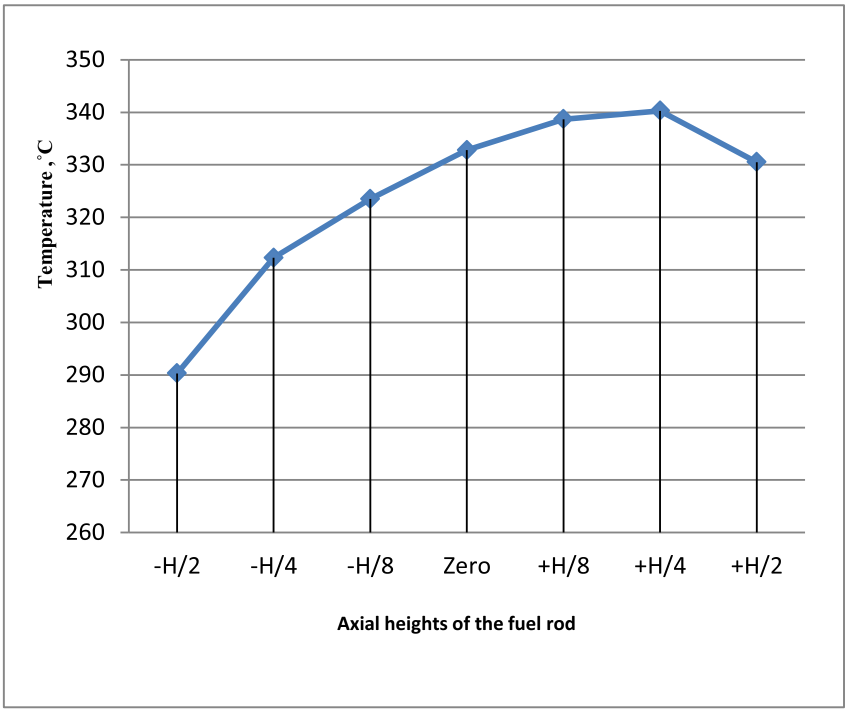

Figure 5. Cladding outer surface temperature axial distribution

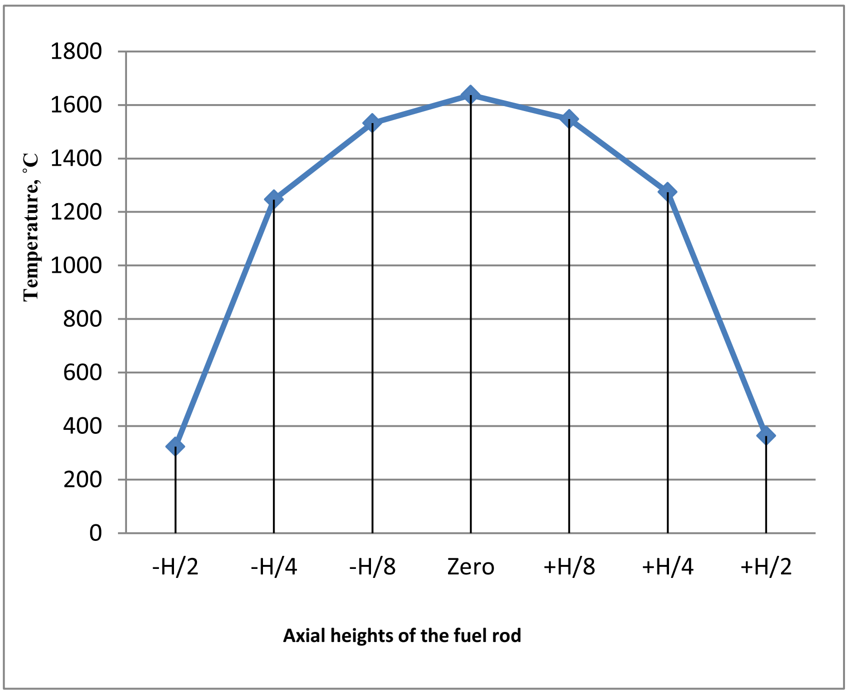

Figure 6. Fuel inner surface temperature axial distribution

3.2 Transient - state

A scenario of occasional malfunction in the primary loop causing an exponential loss of coolant in the hottest channel has been postulated and formulated as follows:

$$\dot{m}\,(t) = \dot{m}_o\, e^{-t/\tau}\,,$$

Where;

τ .. is the time constant,

t .. is the transient time .

According to the IAEA (IAEA, 1992a), the reactor controller shall activate certain protection criteria in case of LOFA. These criteria requires that the reactor must be shut down when the total flow of the coolant passing through it becomes less than 85% of its nominal value. And according to the publications concerning the adopted design safety margins of the reactor core integrity, the following limitations are to be considered throughout the postulated scenario:

a. $T_{c,max}$ < 352.15 °C (at which nucleate boiling occurs).

b. $T_{f,max}$ < 1883.5 °C.

3.2.1 Slow Loss of Flow transient (SLOFT), τ = 250 s.

The safety systems of the reactor are assumed to be not working during the SLOT scenario and so, the peak cladding and fuel temperatures are shown in the following figures:

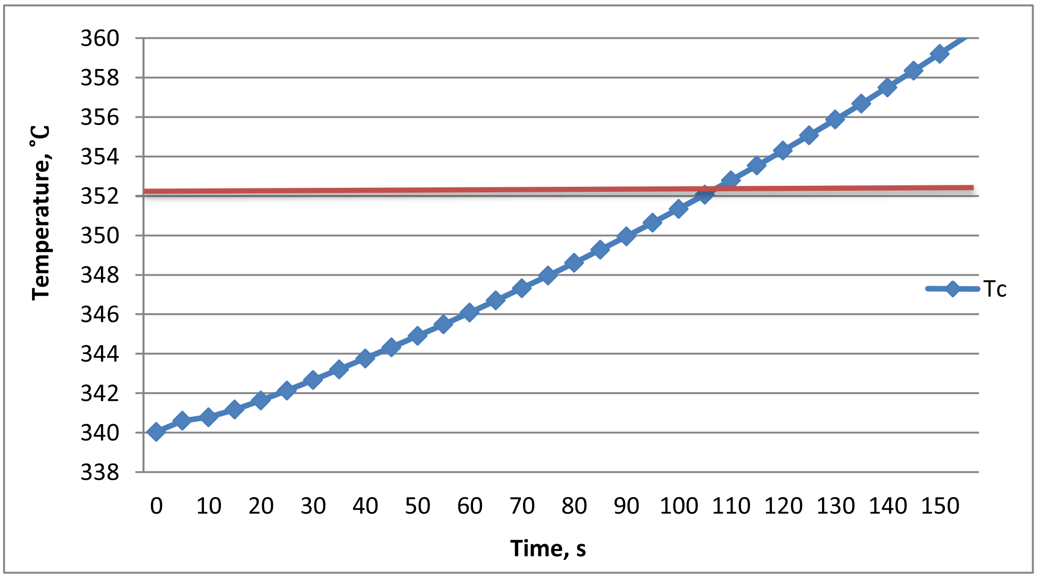

Figure 7. maximum clad outer surface temperature at τ = 250 s (SLOFT)

Through the SLOFT scenario the maximum clad outer surface temperature that located at a distance about 83 cm from the center of the fuel rod to be not exceeding 352˚ C (625 K) is shown to be occurred after about 110 s from the initiation of the transient as shown in figure 7.

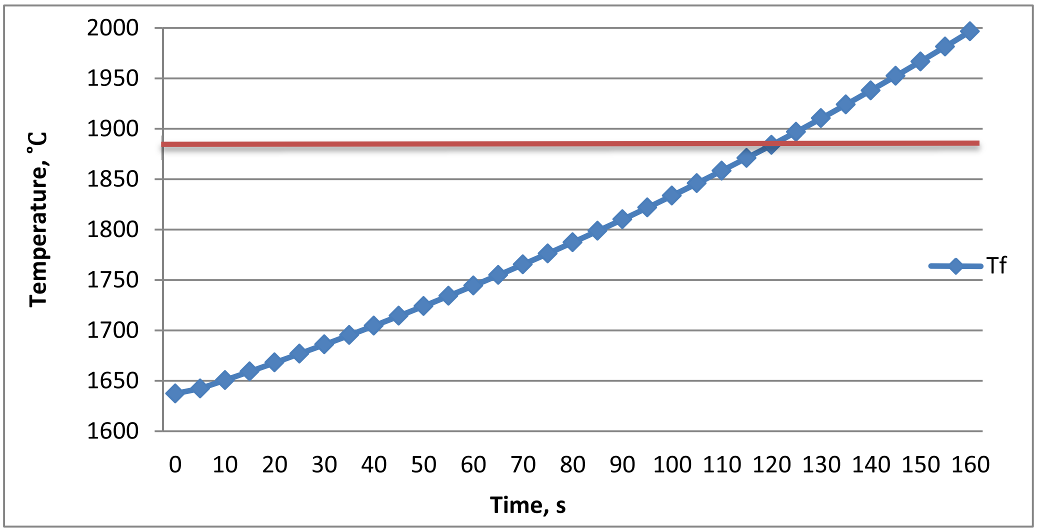

Figure 8. Maximum fuel inner surface temperature at τ = 250 s (SLOFT)

As shown in figure 8 and according to the references that determine the maximum fuel inner surface temperature that located at a distance about 1.5 cm from the center of the fuel rod to be not exceeding 1883.5˚ C (2156.5 K) is shown to be occurred after about 120 s from the initiation of the transient.

There is no doubt that maximum fuel temperature is a very important parameter for code validation because of that Fuel temperature responds to many effects that takes place during irradiation, such as fuel conductivity degradation, fuel densification, swelling, relocation, and other phenomena affecting the gap conductance value.

3.2.2 Fast loss of flow transient (FLOFT), τ = 10 s.

The safety systems of the reactor are assumed to be not working during the SLOT scenario and so, the peak cladding and fuel temperatures are shown in the following figures:

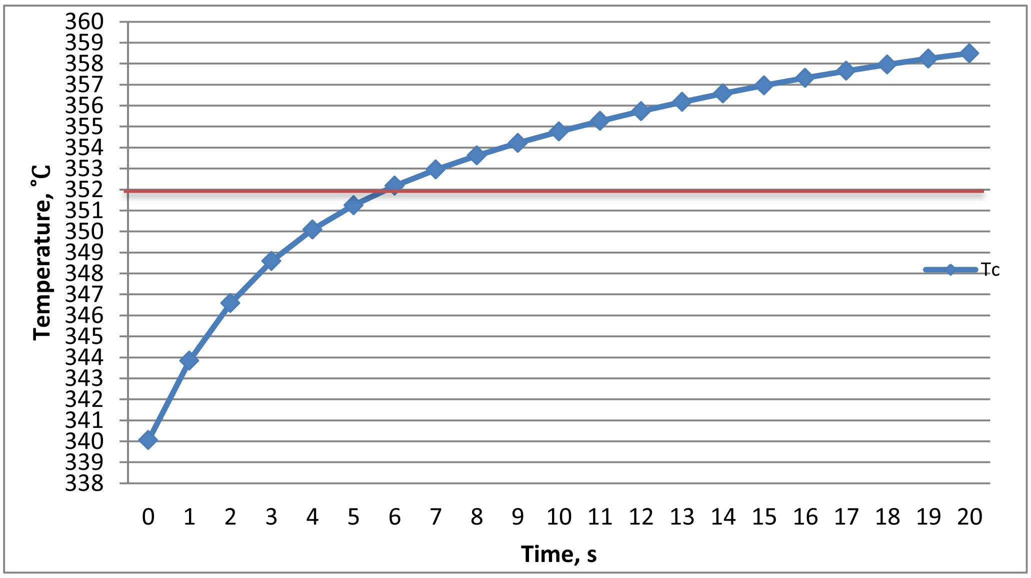

Figure 9. Maximum clad outer surface temperature at τ = 10 s (FLOFT)

According to the references that determine the maximum clad outer surface temperature that located at a distance about 83 cm from the center of the fuel rod (the orange color) to be not exceeding 352˚ C (625 K) is shown to be occurred after about 6 s from the initiation of the transient as shown in figure 9.

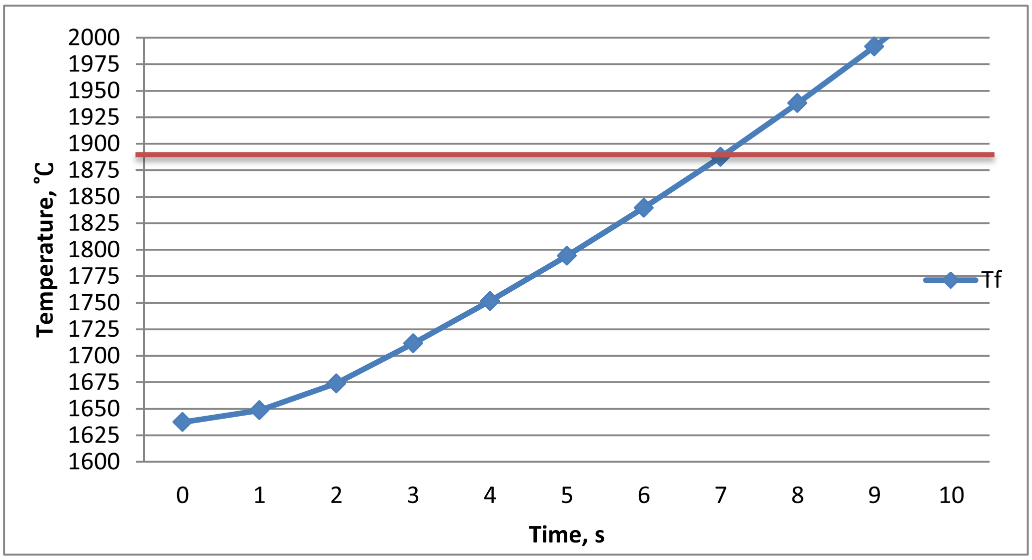

Figure 10. Maximum fuel inner surface temperature at τ =10 s (FLOFT)

On the other side the maximum fuel inner surface temperature is shown to be occurred at about 7 s from the initiation time of the transient that looks like to be reasonable due to the extreme scenario assumed of the rapid loss of flow that of course should resulting in the rapid increase in the fuel inner surface temperature as shown in figure10.

4. Conclusion

Steady-state and the transient state models were solved using finite difference method and MATLAB code. The coolant, cladding and fuel axial temperature distributions for the hottest channel in the reactor core of VVER-1000 NPP were calculated in case of loss of coolant flow event. The results were in good agreement with those found in the literature .The operator and /or the safety systems have always a short period of time for intervention and actuation to preserve the fuel integrity. Calculation prevailed that the available period for the adopted exponential loss of flow scenarios is directly related to the average time constant of the core coolant flow. For the postulated fast loss of flow it was about 6 s which means that the maximum time available for the intervention of both safety systems or the operator is about 7 seconds in order to prevent the melting of the fuel, while in case of the slow loss of flow it was about 120 s.

5. References

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