THE EFFECT OF SUBCOOLED WATER PARAMETERS ON THERMAL-HYDRAULIC CHARACTERISTICS FOR VVER REACTOR

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Abstract.

Prediction of critical heat flux (CHF) using empirical correlations for circular tubes modified for rod bundles with correction factors, is one of established method of CHF evaluation. In this work, an analysis of the thermal-hydraulics of VVER heated core was carried out. A geometrical and thermal analysis of the heated core, including the analysis of the flow rate and mass flux for the assemblies with hot water, analysis of heat transfer densities of the hottest part of the core and proper thermo-fluid analysis of the coolant parameters such as enthalpy, temperature and steam equilibrium quality were carried out. The thermal hydraulic analyses were carried out at the pressures, inlet temperatures and thermal powers of 16.2 MPa, 298.2 $^{\circ}C$ *and 3200* MW_{th}*, 15.7 MPa, 298.2* $^{\circ}C$ *and 3000* MW_{th} *and 12.5 MPa, 262* $^{\circ}C$ *and 1375* MW_{th} *to ascertain the axial changes in thermal parameters of the fuel rod. The critical heat flux (CHF) was predicted using the OKB Gidropress and Levitan-Lantsman CHF correlations for rod bundles under the ranges of parameters suitable for VVER reactor.*

Keywords: thermal, hydraulics, water coolant, fuel rod, nuclear reactor, heat flux, temperature, pressure

1. Introduction

There has been constant improvement and refinement of empirical critical heat flux (CHF) correlations for the past 50 years with more than 100 correlations, developed just for tubes sub-cooled by water [1]. This constant refinement of CHF prediction methods clearly shows that CHF mechanism depends on complex combination of parameters, both thermal and geometrical; as such, no single CHF model can be applied to all CHF conditions of interest [2]. The complex combination of parameters to predict CHF, further increases significantly as additional factors such as transients, non-uniform flux distributions, and asymmetric cross sections are introduced. Considering the complex nature of predicting this phenomenon, experts began collecting experimental data points in an organized format to develop the CHF look-up table [2, 3]. The two popular look-up table are the Bobkov table (2011) and the Groeneveld table (2006). Both of the tables are for round tubes with corrections factor applied to predict CHF for rod bundles. Specifically, the CHF value obtained from the Bobkov table is multiplied by four factors to be applicable to VVER reactor [4]. As noted by Yu. A. Bezrukov et al., [5], with rest respect to a possible power excursion of operating VVER reactor, the basic requirement consists of avoiding any form of boiling crisis, both during normal operation and during transients caused by deviations from normal operating conditions. Increasing reactor thermal power leads to a higher heat flux from the fuel rod surface and a higher temperature at the core outlet. These increase of power factor, causes the departure from the nucleate boiling (DNB) ratio to get smaller, and its evaluated values can be kept at an acceptable level only by making the calculations less conservative or by enhancing heat transfer.

2. Material and Methods

2.1. VVER-1200 Core Design and Calculated Geometrical and Thermal Parameters

For a hexagonal assembly with side *S* and the rods arranged in regular manner, it is possible to deduce certain relationships about the total number of rods. Clearly as we go from the center, the number of coolant regions (channels) increases. In the first ring we have 7 rods; in the second ring we have 19, in the third ring 37 and so on [7].

The VVER-1200 is a reactor with a nominal thermal and electrical powers of 3200 MW_{th} and 1200 MW_e. The core is made up of 163 fuel rod assemblies (FRAs) in hexagonal array with an assembly lattice pitch of 0.235m. A fuel assembly of a typical VVER-1200 reactor consists of 331 rods of which 312 are fuel rods, 18 are control rod guide tubes and 1 central rod, all packed in triangular lattice with a rod to rod pitch of 0.01275 m. The fuel rods contain ceramic fuel of uranium oxide UO_2 pellets, with a cladding material of 98.97 % Zr, 1 % Nb, 0.03 % Hf [8]. Some of the important thermal and geometrical specifications of the core and FRAs are given and calculated in this analysis are listed in Table 1.

Table 1: Main characteristics of VVER-1200 reactor core

Parameter, unit	Value
Thermal Power, [MW]	3200
Coolant Pressure, [MPa]	16.2
Coolant Inlet Temperature, [$^{\circ}$C]	298.2
Coolant Outlet Temperature, [$^{\circ}$C]	328.9
Radial nuclear peaking factor	1.3
Lattice (grid) pitch (absolute), [m]	0.01275
Pitch-to-diameter ratio	1.4
Total number of rods, pc.	331
Number of fuel rods, pc.	312
Number of control rods, pc.	18
Number of central tubes, pc.	1
Number of rods on main diagonal FA	21
Outer diameter of the central tube, [m]	0.0133
The outer diameter of the tube for the control rod, [m]	0.0126
Type of fuel element	rod
The outer diameter of the cladding, [m]	0.0091
The thickness of the cladding, [m]	0.65
Thickness of the gas gap, [m]	0.00013
Diameter of the hole in the fuel pellet, [m]	0.00075
Axial nuclear peaking factor	1.53
Thermal conductivity of fuel, [W m^{-1} K^{-1}]	3.04
Height of the active core, [m]	3.356
Effective height of the core, [m]	3.446
The diameter of the core, [m]	3.356

2.2. Thermal Hydraulics Modeling

Based on the possibility of an escalating thermal crisis in a reactor core due to power excursion during transient, we modeled the thermal-hydraulics analysis of VVER-1200 reactor using the initial design parameters and calculated the thermal-physical parameters of the coolant and the geometrical and thermal parameters of the fuel elements. The Subbotin correlation was used to determine the Nusselt number and hence the heat transfer coefficient of the water coolant with rod surface. We employed the OKB Gidropress [8] and Levitan-Lantsman [9] correlations to predict the critical heat flux considering that they are well established correlations for triangular lattice with relevant correction factors and application ranges for a VVER reactor. The OKB Gidropress empirical correlation for CHF was obtained under conditions as close as possible to the operating conditions of the VVER-1000 and VVER-1200 reactors. It also described experimental data in the ranges given with a standard deviation of about 13.1% and an average algebraic deviation of points from the computed formula of about 1 %. The Levitan-Lantsman correlation [9] was popular in the Soviet Union at the time when the two popular pressurized water reactors (PWRs) CHF correlations were the Westinghouse atomic power division W-3 correlation and the Levitan–Lantsman correlation. While the western PWR used the W-3 correlation for CHF prediction, the Soviet VVER used Levitan-Lantsman correlation. The Levitan-Lantsman CHF correlation has a standard deviation of about ±15 %.

The Design of thermal-hydraulic calculation of a pressurized water reactor VVER-1200 was carried out with the computed and given geometrical and thermal parameters of the fuel, cladding and the entire core in Table 1, and the thermal parameters of the coolant were computed with the aid of IAPWS IF97 Excel Steam Tables.

In our design calculation, the maximum linear heat generation rate was calculated to be 37.27 kW m^{-1} at 16.2 MPa and 3200 MW$_{th}$; 34.9 kW m^{-1} at 15.7 MPa and 3000 MW$_{th}$ and 16 kW m^{-1} at 12.5 MPa and 1375 MW$_{th}$. This trend is consistent, provided that the coolant inlet and outlet conditions are in phase with pressure and power drop. In our case the inlet coolant temperature is 298.2 $^{\circ}$C for both pressures of 16.2 MPa and 15.7 MPa, while that of 12.5 MPa is 262 $^{\circ}$C.

$$\dot{Q}_{fuel}(z)=\int_{-0.5H}^{z} q_l'(z)\cdot dz=\frac{q_{l,\max}'\cdot H_e}{\pi}\cdot\left[\sin\left(\frac{\pi\cdot z}{H_e}\right)+\sin\left(\frac{\pi\cdot H}{2\cdot H_e}\right)\right], \quad (1)$$

$$q_l'(z)=q_{l,\max}'\cdot\cos\left(\frac{\pi\cdot z}{H_e}\right). \quad (2)$$

The equilibrium quality is given as:

$$x_{eq}(z)=\frac{\dfrac{q_{l,\max}\cdot H_e}{\dot{m}_{fr}\cdot\pi}\left[\sin\left(\dfrac{\pi\cdot z}{H_e}\right)+\sin\left(\dfrac{\pi\cdot H}{2\cdot H_e}\right)\right]+\left(h_{in}-h_f\right)}{h_g-h_f}, \quad (3)$$

where, h_g – enthalpy of coolant at saturated vapor; h_f – enthalpy of coolant at saturated liquid.

The operating heat flux density $q_f''(z)$:

$$q_f''(z)\cdot\pi\cdot d_{fr}\cdot H=q_l'(z)\cdot H. \quad (4)$$

The Subbotin correlation for water cooled rod bundle is given as:

$$\mathrm{Nu}=A_1\cdot\mathrm{Re}^{0.8}\cdot\mathrm{Pr}^{0.4}, \quad (5)$$

where $A_1=0.0165+0.02\cdot\left(1-\dfrac{0.91}{\left(\dfrac{p_{fr}}{d_{fr}}\right)^2}\right)\cdot\left(\dfrac{p_{fr}}{d_{fr}}\right)^{0.15}$.

OKB Gidropress critical heat flux correlation

$$q_{cr}''=0.795\cdot\left(1-x_{eq}\right)^n\cdot(G)^m\cdot(1-0.0185\cdot p), \quad (6)$$

where, $m=0.311\cdot\left(1-x_{eq}\right)-0.127$, $n=0.105\cdot p-0.5=0.105\cdot 16.2-0.5=1.201$, p is pressure, [MPa], G is the mass flux, [kgm^{-2}s^{-1}]

Levitan-Lantsman critical heat flux correlation:

$$q''_{cr}\left(0.008, G, p, x_{eq}\right) = \left[10.3 - 7.8\left(\frac{p}{98}\right) + 1.6\left(\frac{p}{98}\right)^2\right] \cdot \left(\frac{G}{1000}\right)^{1.2\left\{\left[0.25(p-98)/98\right]-x_{eq}\right\}} \cdot e^{-1.5\cdot x_{eq}} \tag{7}$$

Where, $29.4 \le p \le 196$ [bar]; $750 \le G \le 5000$ [kg m^{-2} s^{-1}]

$$q''_{cr}\left(d_{fr}, (G), p, x_{eq}\right) = q''_{cr}\left(0.008m, (G), p, x_{eq}\right)\left(\frac{0.008}{d_{fr}}\right)^{0.5}, \tag{8}$$

where, x_{eq} is relative enthalpy of the water coolant; G is mass flux, [kg m^{-2} s^{-1}], p is the pressure, [bar]; d_{fr} is the outer diameter of the fuel cladding, [m].

Departure from nucleate boiling ratio (DNBR):

$$DNBR(z) = \frac{q''_{cr}(z)}{q''_{f}(z)}. \tag{9}$$

The coolant temperature is given as:

$$t_{coolant}(z) = \frac{(t_{in} + t_{out})}{2} + \frac{(t_{out} - t_{in})}{2\sin\left(\frac{\pi H}{2 \cdot H_e}\right)} \sin\left(\frac{\pi z}{H_e}\right). \tag{10}$$

The temperature of the outer surface of the cladding of a fuel rod is given as:

$$t_{out,clad} = t_{coolant}(z) + \frac{q'_l(z)}{\pi \cdot d_{fr} \cdot \alpha_{coolant}}. \tag{11}$$

The temperature of the inner surface of the cladding of a fuel rod is given as:

$$t_{in,clad}(z) = t_{out,clad}(z) + \frac{0,94 \cdot q'_l(z)}{\pi \cdot \left(d_{fr} - \delta_{clad}\right)} \cdot \frac{\delta_{clad}}{\lambda_{clad}}. \tag{12}$$

Temperature of the outer surface of the fuel pellets is given as:

$$t_{f,surf} = t_{in,clad}(z) + \frac{0,94 \cdot q'_l(z)}{\pi \cdot \left(d_{fr} - \delta_{clad} - \delta_{gg}\right) \cdot \alpha_{gg}}. \tag{13}$$

Temperature in the center of the fuel pellets is given as:

$$t_{centerline}(z) = t_{f,surf}(z) + \frac{0,94 \cdot q'_l(z)}{4 \cdot \pi \cdot \lambda_{fp}} \cdot \left(1 - \frac{2 \cdot d^2_{annular}}{d^2_{fp} - d^2_{annular}} \cdot \ln\frac{d_{fp}}{d_{annular}}\right). \tag{14}$$

3. Result and Analysis

Thermal energy is emitted unevenly by the volume of the nuclear reactor core. Fuel assemblies with the maximum heat release are the most important elements of the reactor design and are given special attention. Consider the distribution of heat release and temperatures in FA height in nominal operating mode of reactor.

The average thermal power of a fuel element in the core at 16.2 MPa, amounted to about 41 kW, this value when multiplied by the total number of fuel rod (50856) and the axial peaking factor (1.53) gives approximately 3190 MW_{th}. Repeating the same process for 15.7 MPa and 12.5 MPa, amounted to 2980 MW_{th} and 1366 MW_{th} respectively. The thermal power is a function of heat generated by length of the rod, it is also worthy of note that the thermal power depends largely on the coolant ability to extract heat from the core of the reactor. Factors like coolant flow rate and inlet temperature influences heat extraction by the coolant. This is the reason why adjustment of flow rate impact much on the thermal power. At 3200 MW_{th}, the flow rate was 17721 kg s^{-1}, at 3000 MW_{th}, the flow rate was 16526 kg s^{-1} and at 1375 MW_{th}, the flow rate was 7499 kg s^{-1}.

According to Eq. (2) and the distribution graph in Fig. 1, the maximum fuel rod linear heat generation rate (LHGR) at normal VVER-1200 operating condition was 37.27 kW m^{-1} which is below the design value of 42 kW m^{-1}, the reason for this arises from thermal hydraulics core design used in the calculation. As the power dropped to 3000 MW_{th}, the linear heat generation rate was calculated to be 34.95 kW m^{-1}, at 1375 MW_{th}, the linear heat generation rate was calculated to be 16 kW m^{-1}.

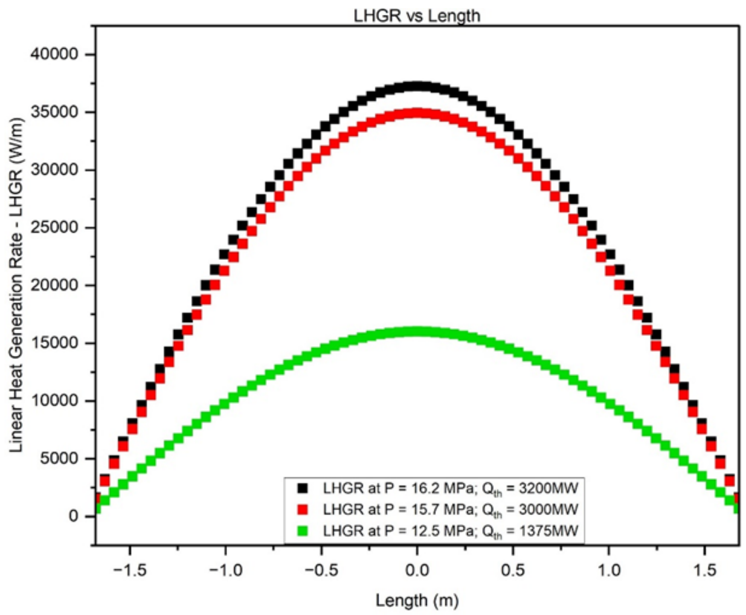
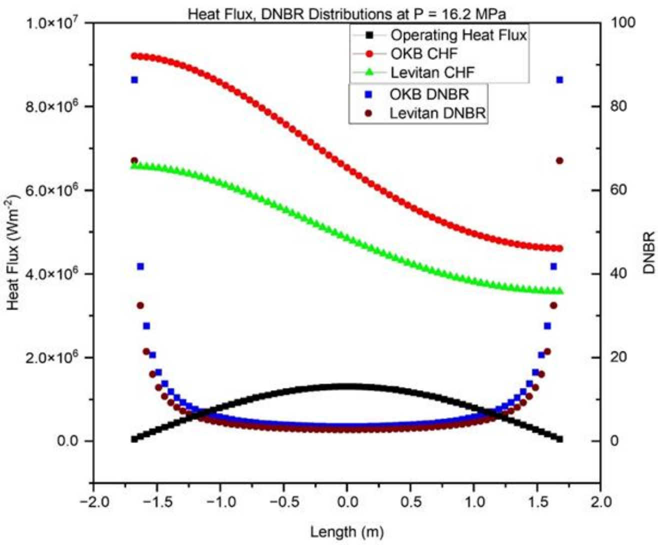


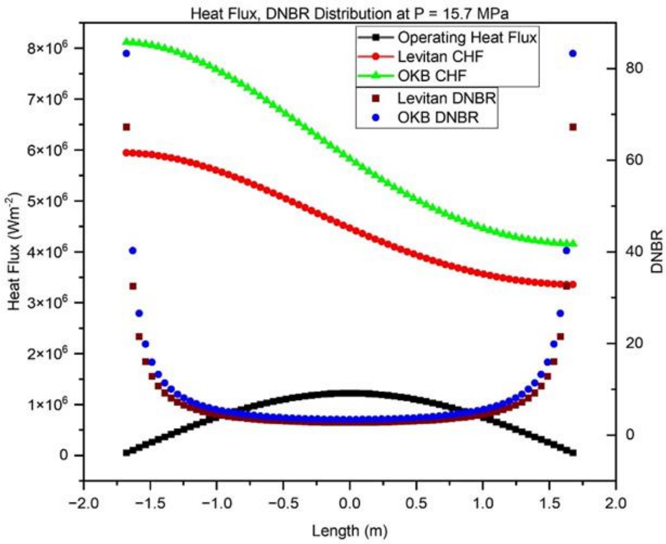
Fig 1: Linear heat generation rate on core length at 16.2MPa, 15.7MPa and 12.5MPa

From Eqs. (4-8) and the heat flux, CHF and DNBR distributions in Fig 2 (a-c), the critical heat flux correlations used for the analysis were selected based on their accuracy for predicting departure from nucleate boiling under high pressure, high heat flux, high mass flux and low equilibrium quality. Both the OKB Gidropress and Levitan-Lantsman correlations are functions of pressure, mass flux and equilibrium quality, which replicates the functions of CHF lookup tables. At 16.2MPa in Fig. 2(a), the OKB Gidropress correlation predicted the minimum departure from nucleate boiling ratio (MDNBR) to be 3.52. While the Levitan-Lantsman correlation predicted the minimum departure from nucleate boiling ratio (MDNBR) to be 2.73, at 15.7 MPa in Fig. 2(b), the OKB Gidropress correlation predicted the MDNBR to be 3.39, while the Levitan-Lantsamn correlation predicted it to be 2.73, and at

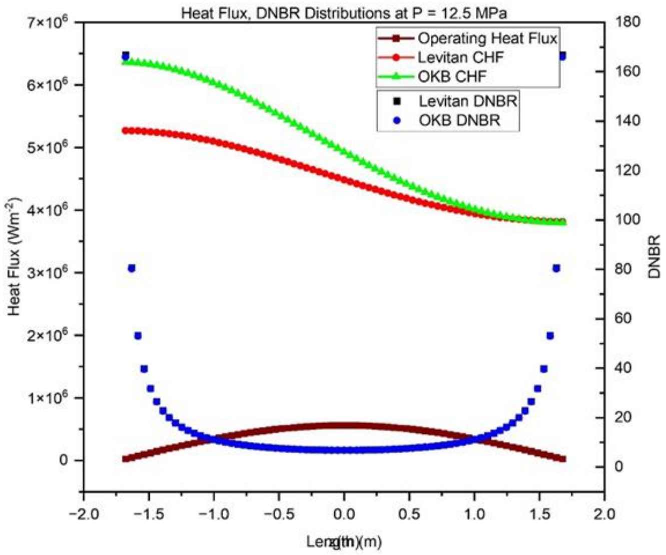
12.5 MPa in Fig. 2(c), the OKB Gidropress correlation predicted the MDNBR to be 6.8 same for Levitan-Lantsman correlation. The results of the CHF prediction showed that the Levitan-Lantsman correlation predicted the MDNBR better with an estimated value of about 2.74 at the VVER-1200 operational pressure of 16.2 MPa, The Levitan-Lantsman Correlation has been known to predict DNB for sub-cooled flow boiling with great accuracy, hence it was used for this analysis.



(a)



(b)



(c)

Fig 2: Heat flux, CHF and DNBR distribution at (a) 16.2MPa, (b) 15.7MPa and (c) 12.5MPa

In the temperature distribution graphs of Fig. 3 (a-f), represented by Eqs. (9-14), coolant temperature, surface temperatures showed the normal trend of a typical reactor temperature, with increasing maximum point shift towards the center as we move layer by layer towards the centerline point of the fuel element. The computed outer surface cladding temperature was within 342 $^{\circ}$C at 16.2 MPa, 340 $^{\circ}$C at 15.7 MPa and 305 $^{\circ}$C at 12.5 MPa, cladding temperatures were far from the crisis zone where maximum temperature of the outer surface of the fuel cladding should not exceed the maximum for zirconium alloys in the water coolant (360–365 $^{\circ}$C)

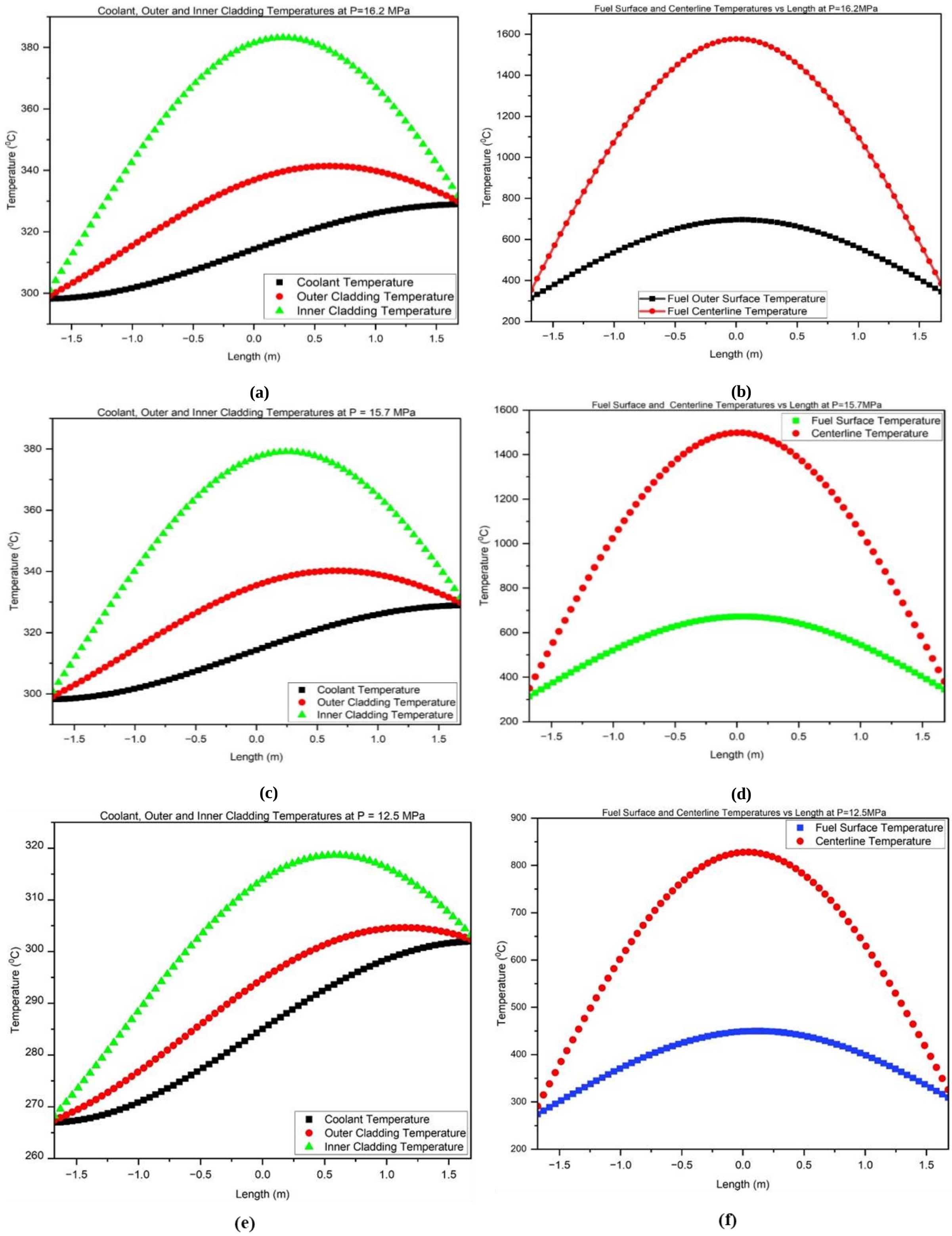


Fig. 3: (a) Coolant, outer and inner cladding temperatures at 16.2 MPa; (b) Fuel surface and centerline temperatures at 16.2MPa; (c) Coolant, outer and inner cladding temperatures at 15.7 MPa; (d) Fuel surface and centerline temperatures at 15.7 MPa; (e) Coolant, outer and inner cladding temperatures at 12.5 MPa; (f) Fuel surface and centerline temperatures at 12.5 MPa

4. Conclusion

The computed average linear heat flux was within the operating linear heat flux of 15.7–19.0 kW m^{-1} for a typical pressurized water reactor (PWR) [10] at 16.2MPa. The reason for the slight increase in the linear heat flux compared to the design value of 16.78 kW m^{-1} for a VVER-1000/1200 was due to the dimension of our reactor core, we designed our core to have a height to diameter ratio of 1.

The Levitan-Lantsman correlation with prediction accuracy of ±15%, predicted the minimum departure from nucleate boiling ratio to be 2.73, 2.74 and 6.8 at 16.2 MPa, 15.7 MPa and 12.5 MPa respectively. The OKB Gidropress correlation with prediction accuracy of ±13.1% on the other hand predicted the MDNBR to be 3.52, 3.39 and 6.8 at 16.2 MPa, 15.7 MPa and 12.5 MPa respectively. In most literatures, the predicted MDNBR for VVER reactors ranges from 1.61 to 2.62 [11]. Therefore, we can say that under normal operational condition, no CHF occurred in the reactor core, however, CHF can be triggered by power excursion, increase in coolant inlet temperature and drop in coolant pressure or flow rate. These CHF triggering mechanisms are highly monitored during reactor operation.

Nomenclature

A_1	- Subottin correlation constant	q''_f	- heat flux [Wm^{-2}]
$DNBR$	- departure from nucleate boiling ratio	q''_{cr}	- critical heat flux [Wm^{-2}]
$d_{annular}$	- diameter of the hole in the fuel pellet [m]	Re	- Reynolds number [$Re = G \cdot D_h / \mu$]
d_{fp}	- diameter of fuel pellet [m]	$t_{coolant}$	- temperature of liquid [°C]
d_{fr}	- diameter of fuel rod [m]	t_{sat}	- saturation temperature [°C]
G	- mass flux [$kgm^{-2}s^{-1}$]	t_{in}	- inlet temperature [°C]
H	- height of active cylindrical core [m]	t_{out}	- outlet temperature [°C]
H_e	- effective height of the core [m]	$t_{in,clad}$	- inner clad temperature [°C]
h_g	- enthalpy at saturated vapor [$kJkg^{-1}$]	$t_{out,clad}$	- outer clad temperature [°C]
h_{fg}	- latent heat of vaporization [$kJkg^{-1}$]	$t_{f,surf}$	- fuel surface temperature [°C]
h_f	- enthalpy of liquid at saturation [$kJkg^{-1}$]	$t_{centerline}$	- fuel centerline temperature [°C]
$\dot{m}_{fr}$	- coolant flow rate along fuel rod [kgs^{-1}]	x_{eq}	- equilibrium quality
N	- total number of fuel rods in a lattice	z	- channel axial coordinate [m]
p	- coolant pressure [MPa]	*Greek Symbols*	
p_{fr}	- fuel rod pitch [m]	$\alpha_{coolant}$	- heat transfer coefficient [$Wm^{-2}K^{-1}$]
Pr	- Prandtl [$Pr = \mu \cdot c_p / k_{coolant}$]	α_{gg}	- gas gap conductance, [$Wm^{-2}K^{-1}$]
q''_f	- heat flux [Wm^{-2}]	δ_{clad}	- cladding thickness [m]
q'_l	- linear heat flux of fuel rod [Wm^{-2}]	δ_{gg}	- gas gap thickness [m]
$q'_{l,\max}$	- maximum linear heat flux of rod [Wm^{-1}]	η	- number fuel rod rings

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Paper submitted: 30 March 2024
Paper revised: 18 April 2024
Paper accepted: 19 April 2024