International Research Journal of Engineering and Technology (IRJET) e-ISSN: 2395-0056
Volume: 08 Issue: 07 | Jul 2021 www.irjet.net p-ISSN: 2395-0072

TEMPERATURE PROFILE AND THERMAL PERFORMANCE OF BOILING WATER REACTOR BASED POWER PLANT

Govind Maheshwari[1], Vineet Kumar Saraswat[2], Jitendra Kumar[3], Zafar Alam[4*], Adnan Hafiz[5]

[1-3]UG Student, Mechanical Engineering Department, Aligarh Muslim University, Aligarh, India. 202002
[4]M.Tech Student, Mechanical Engineering Department, Aligarh Muslim University, Aligarh, India. 202002*
[5]Associate Professor, Mechanical Engineering Department, Aligarh Muslim University, Aligarh, India. 202002

---***---

Abstract - *A recent BWR fuel assembly includes of 37 fuel rods and there are up to roughly 800 assemblies in a reactor. Reactor core size, desired reactor power output, and reactor power density are the consideration are based on the number of fuel assemblies in a specific. In our analysis capacity of nuclear power plant is approximately 1465 MW. Nuclear reactor core is generally made up of fuel elements in cylindrical shape and contains fuel pellets, helium gas gap and cladding material. In this paper our ultimate goal was to analyze the temperature drop across the fuel element, from the center of the fuel where the maximum temperature occurs, to the gas gap and then finally to the cladding surface. The physical and thermal properties of the fuel element are known. The Nuclear fuel element used in this analysis was uranium dioxide, which is the standard fuel element for Light Water Reactors. The thickness of the helium gas gap is kept small because of its very low thermal conductivity, hence it is not very good in conducting heat between two surfaces.*

Key Words**: fuel pellets, cladding, water reactor, thermal conductivity, Temperature drop**

1. INTRODUCTION

In designing of a nuclear power plant, calculation of the power output from the reactor core and its removal by the coolant are very important. Coolant is circulated around the core and heat flows from the centre of fuel rod to the coolant. The study of heat transfer of fuel element along the radial direction is very important for the safety of a nuclear reactor. This is due to the fact that when heat is properly transferred between the layers of materials that make up the fuel element, the integrity of the fuel assembly structure would be compromised. This is the reason why the thermal conductivities, specific heat capacities and densities of these layers of materials that make up the fuel element are highly considered when choosing them. In modern nuclear power plants, Uranium dioxide is one of the most widely used fuel, but its low thermal conductivity is a real problem. The low thermal conductivity usually aid the formation of cracks in the pellet, however this negativity is overcome by its high melting point [1]. Hino et al. [2] proposed a customary dissemination computation utilizing atomic constants arranged by two-dimensional grid estimations was performed by Hitachi, although the computation utilizing atomic constants arranged by three-dimensional approximations and hub intermittence factors was achieved by the College of Michigan to give a more modern treatment of the pivotal heterogeneity. Wu et al. [3] showed the center plan and execution attributes of the Novel Measured Reactor (NMR-50), a 50-MW(electric) little particular reactor. With regular dissemination cooling, NMR-50 is a bubbling water reactor and two layers of detached wellbeing frameworks that empower the reactor to withstand delayed station power outage and loss of extreme warmth sink mishaps. Configuration concentrates with the CASMO-4 cross section code and coupled neutronics and warm water driven center investigations with the PARCS and RELAP5 codes, a primer NMR-50 center plan has been created to meet the 10-year cycle length with a normal fuel advancement of 4.75 wt% and a most extreme enhancement of 5.0 wt%. Mieloszyk et al.[4] proposed primer appraisal of a normal RBWR-Th fuel pole shows the fuel temperature to stay under 1450 K and the fission gas release (FGR) to stay underneath 7%. Be that as it may, in view of the low free gas volume of the RBWR-Th poles, the plenum pressure is exceptionally delicate to FGR and is demonstrated to be equipped for surpassing the coolant pressure. Of more concern is the high cladding hydrogen content that outcomes from the speed increase of hydrogen pickup at generally low burnups, which is brought about by the high quick neutron fluence on the cladding in the RBWR-Th. Another cladding material, GNF-Ziron, from Worldwide Atomic Powers (GNF) offers a likely answer for this test by postponing the speed increase of the hydrogen pickup. Sukjai et al. [5] proposed an examination recommends that due to high pivotal cresting factors and somewhat level force history, fuel temperature is essentially higher in fissile zones of the RBWR-TB2 prompting higher splitting gas delivery and cladding deformity. The higher neighborhood fuel burnup in fissile zones of RBWR-TB2 additionally prompts huge however worthy fuel expanding, sped up cladding oxidation, and PCMI. To help future plan enhancements, affectability investigations on significant fuel plan boundaries are additionally performed. Atkinson et al. [6] Recommended high constancy strategies through Monte Carlo/Sub-channel examination contrasted with modern techniques for cross-area/nodal investigation utilizing the High level Bubbling Water Reactor as a contextual analysis. Their examination was picked because of the difficulties in demonstrating two stage stream and the significant degrees of heterogeneity inside the fuel get together plan. They examined how to carry out such a methodology, from a base up technique, by breaking down each phase of the demonstrating cycle. Hussain et al. [7] compare the performance of evaporatively cooled and steel wire mesh condenser and improve the heat rejection rate with less energy utilization. Khan et al. [8] involves the employment of nano fluid to reduce the temperature of PV panel. They used Zinc nanoparticles in different base fluids and thermal efficiency was observed to be 31% then the conventional system. Hussain et al. [9] analysed the transient heat transfer behaviour of cylindrical food product and suggested the optimum value of precooling rate of cylindrical food product which is 3.8 m/s air inlet velocity. Alam et al. [10] signified the predetermined heat transfer condition of a multistorey building using heat transfer approach. Hussain et al. [11] showed the analysis of solid desiccant dehumidifier and calculated the three diffusion coefficient. On the basis of which suggested a best model for the same.

2. METHODOLOGY

By embedding's or pulling out control poles and by changing the water move through the reactor center are two strategies by Reactor power is controlled:

Situating (pulling out or embedding's) control poles is the ordinary strategy for governing force when firing up a BWR., neutron retention diminishes in the control material and expansions in the fuel, as control bars are removed, power increments of reactor takes place. Reactor power diminishes as control bars are embedded, neutron assimilation expansions in the control material and diminishes in the fuel. In BWR the control bars are embedded from beneath to give a more homogeneous dissemination of the force: fume development, making the neutron balance less productive and the parting likelihood lower because in the upper side the thickness of the water is lower. In ordinary activity, the control bars are simply conditioned to keep a homogeneous force conveyance in the reactor and repay the utilization of the fuel, while the force is controlled concluded the water stream.

Changing (expanding or diminishing) the progression of water through the center is the ordinary and advantageous strategy for controlling force from roughly 30% to 100% reactor power. While working on the supposed "100% bar line," force might be differed from roughly 30% to 100% of appraised power by changing the reactor distribution framework stream by fluctuating the speed of the distribution siphons or tweaking stream control valves.

2.1 Temperature Profile along radial direction in steady state heat transfer in nuclear fuel element

Formulation of Analytical Result

The equation of heat conduction in cylindrical coordinate for the steady state condition can be written as -

$$\frac{1}{r}\frac{d}{dr}\left(r\frac{dT}{dr}\right)+\frac{q}{k}=0$$

Where, q=heat generation

k=thermal conductivity of material,

T= temperature distribution along radial direction,

R=radial distance

Boundary conditions

(a). At the center of fuel pellet temperature is constant, hence temperature gradient is zero at the centerline.

$$\left(\frac{dT}{dr}\right)_{r=0}=0$$

(b). At the pellet outer diameter and at the gas gap inner diameter, the heat flux is constant or the linear heat density is constant, therefore we get,

$$K_{fe}\left(\frac{dT}{dr}\right)_{r=R_{fp}}=K_{gg}\left(\frac{dT}{dr}\right)_{r=R_{fp}}$$

(c). At outer diameter of the gas gap and at the inner surface of the cladding, the heat flux is also assumed to be constant,

$$K_{gg}\left(\frac{dT}{dr}\right)_{r=R_{gg}}=K_{cl}\left(\frac{dT}{dr}\right)_{r=R_{gg}}$$

(d). At the boundary between pellet and gas gap, the temperature is the same,

$$T_{fp}=T_{gg}\qquad At\ r=R_{fp}$$

(e). Between the outer surface of the cladding and the coolant, the thermal flux depends on the temperature difference between cladding and coolant, and the heat transfer coefficient of coolant is given as -

$$K_{cl}\left(-\frac{dT}{dr}\right)_{r=R_{ci}}=h(T_{cl}-T_{cool})$$

(f). At the boundary between the gas gap and the cladding, temperature is the same,

$$T_{gg}=T_{cl}\qquad At\ \ r=R_{gg}$$

Differential temperature distribution equation

(a) For fuel pellet is given by

$$\frac{1}{r}\frac{d}{dr}\left(r\frac{dT}{dr}\right)+\frac{q}{K_{fp}}=0$$

(b) For gas gap is given by

$$\frac{d}{dr}\left(r\frac{dT}{dr}\right)=0$$

(c) For cladding is given by

$$\frac{\mathrm{d}}{\mathrm{dr}}\left(\mathrm{r}\frac{\mathrm{dT}}{\mathrm{dr}}\right)=0$$

Solution of above differential equations

(a) For fuel pellet

$$T = -\frac{qr^2}{4K_{fp}} + C_1 \ln r + C_2$$

(b) For gas gap

$$T = C_4 \ln r + C_3$$

(c) For cladding

$$T = C_5 \ln r + C_6$$

After putting boundary conditions

(a) For fuel pellet

$$T = -\frac{qr^2}{4K_{fp}} + \frac{qR_{fp}^2}{4K_{fp}} + \frac{qR_{fp}^2}{2hR_{cl}} + \frac{qR_{fp}^2}{2K_{cl}} \ln\frac{R_{cl}}{R_{gg}} + \frac{qR_{fp}^2}{2K_{gg}} \ln\frac{R_{gg}}{R_{fp}} + T_{cool}$$

(b) For gas gap

$$T = \frac{qR_{fp}^2}{2hR_{cl}} + \frac{qR_{fp}^2}{2K_{cl}} \ln\frac{R_{cl}}{R_{gg}} + \frac{qR_{fp}^2}{2K_{gg}} \ln\frac{R_{gg}}{r} + T_{cool}$$

(c) For cladding

$$T = \frac{qR_{fp}^2}{2hR_{cl}} + \frac{qR_{fp}^2}{2K_{cl}} \ln\frac{R_{cl}}{r} + T_{cool}$$

Table -1: Shows the Assumed Parameters

Parameter	Symbol	Value
Volumetric heat density of fuel	q	428011969.6 W/m^3
Heat transfer coefficient	h	40KW/Km^2
Radius of fuel pellets	R_{fp}	5.785 mm
Outer radius of cladding	R_{cl}	6.45 mm
Outer radius of gas gap	R_{gg}	5.905 mm
Fuel pellet thermal conductivity	K_{fp}	3.5W/mK
Gas thermal conductivity	K_{gg}	0.15W/mK
Thermal conductivity of cladding	K_{cl}	18W/mK
Length of fuel rod	L	5560 mm
Coolant temperature	T_{cool}	286°C
Linear heat source	q_L	450W/cm

Analytical Result

Temperature Distribution

(a) For fuel pellets

$$T = 2352.311 - 30572283.54\, r^2$$

where, $0 \leq r \leq 0.005785$

(b) For gas gap

$$T = 348.885 + 47746.483 \ln\frac{0.005905}{r}$$

where, $0.005785 \leq r \leq 0.005905$

(c) For cladding

$$T = 313.76 + 397.887 \ln\frac{0.00645}{r}$$

where, $0.005905 \leq r \leq 0.00645$ [12-13]

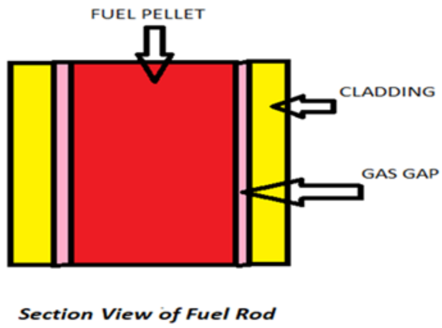

Fig -1: Shows section view of fuel rod

2.2 Mathematical Modelling Nuclear Reactor

Figure 2 & 3 shows layout and T-S diagram of BWR taken into consideration in this paper. Following equations are used for energy balance on the components present in nuclear reactor [14-15].

Heat received from nuclear reactor

$Q = m_0 (h6 - h5)$

m_0= mass flow rate of saturated steam after steam separator (kg/s); h_6 = enthalpy at reactor outlet (kJ/kg);
h_5 = enthalpy at reactor inlet (kJ/kg)

Heat received from super heater

$Q\,s = m_0(h_7 - h_6)$

M_o = mass flow rate of saturated steam after steam separator (kg/s); h_6 = enthalpy at reactor outlet (kJ/kg); h_5 = enthalpy at super heater outlet (kJ/kg)

$h_7 - h_6 = c_p(T_7 - T_6)$

c_p= 4.187 kJ/kgK

Work output from steam turbine

$W_{out} = m_o(h_7 - h_8) + (m_o - m_1)(h_8 - h_1)$

h_1 = enthalpy at condenser inlet; m_1= mass flow rate of water to OFWH (kg/s)

Energy balance across condenser

$Q_{cond} = (m_o - m_1)(h_1 - h_2)$

h_2= enthalpy at condenser outlet; h_1 = enthalpy at condenser inlet

Work done by condenser pump

$W_{P1} = (h_3 - h_2) = v_2(p_3 - p_2)$

v_2= specific volume at condenser outlet; h_3= enthalpy at condenser pump outlet; h_2= enthalpy at condenser outlet

Work done by feed water pump

$W_{P2} = (h_5 - h_4) = v_4(p_5 - p_4)$

V_4= specific volume at feed water pump inlet; h_5=enthalpy at reactor inlet;

h_4=enthalpy at feed water pump inlet

Energy balance across open feed water heater

$m_1 h_8 + (m_o - m_1)h_3 = m_0 h_4$

h_8= enthalpy at OFWH inlet; h_4=enthalpy at feed water pump inlet;

h_3= enthalpy at condenser pump outlet;

m_o= mass flow rate of saturated steam after steam separator(kg/s);

m_1= mass flow rate of water to OFWH (kg/s)

$$m_o = 1737 \left(\frac{kg}{s}\right)$$

Total mass flow rate, m = 13570kg/s

Model Assumption

Table -2: Shows Main Parameters taken into consideration

Main parameters	Value
Outlet temperature of super heater	450 ° C
Reactor and super heater pressure	70bar
Reactor outlet temperature	285.8° C
OFWH pressure	20 bar
OFWH temperature 212.4° C	212.4° C

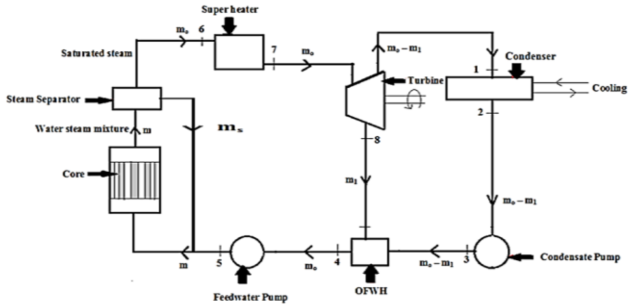

Fig -2: Layout of Nuclear Power Plant with BWR

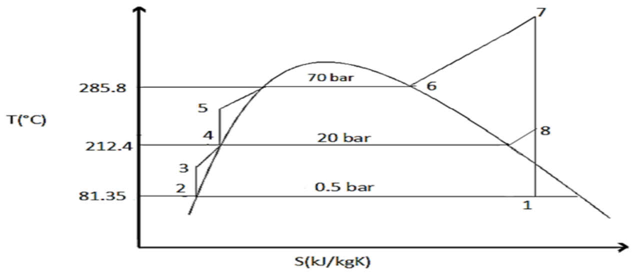

Fig -3: T-S Diagram of Nuclear Power Plant

3. RESULTS

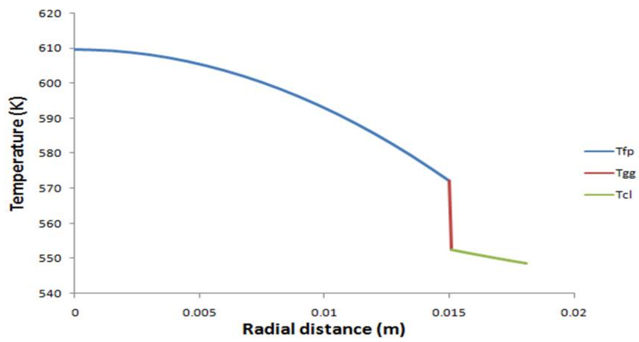

Fig -4: Variation of Temperature in Radial direction in the Fuel Pellets

Tfp- Temperature distribution in Fuel pellets
Tgg- Temperature distribution in gas gap
Tcl- Temperature distribution in Cladding

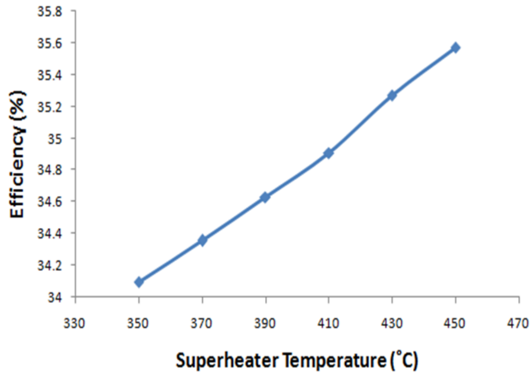

Fig -5: Shows Variation of Efficiency of plant with Super heater Temperature

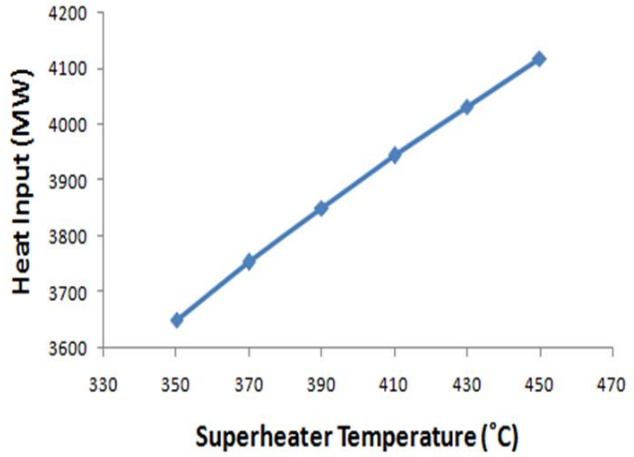

Fig -6: Shows Variation of heat input with Super heater temperature

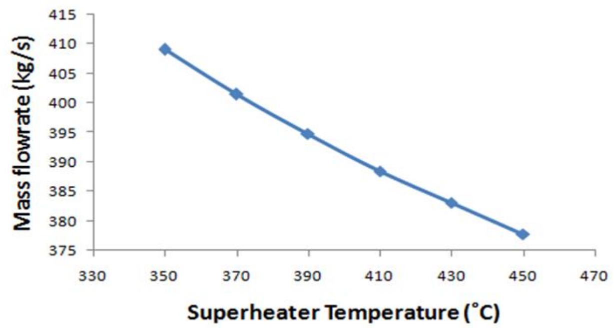

Fig -7: Shows Variation of Mass flow rate in OFWH with Super heater Temperature.

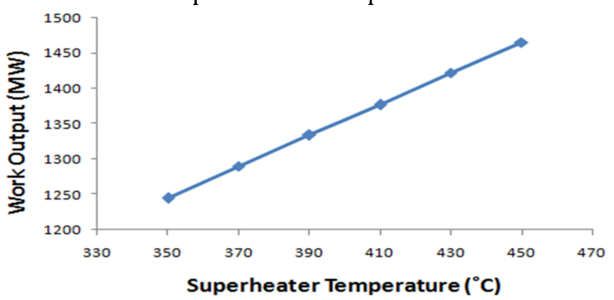

Fig -8: Shows Variation of Work output with Super heater Temperature.

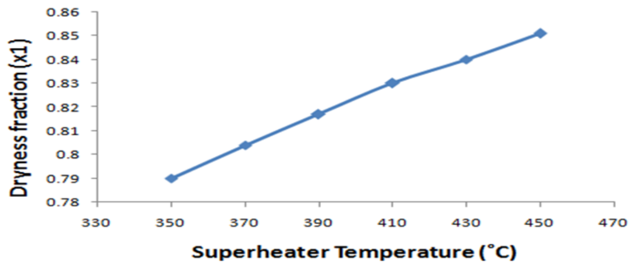

Fig -9: Shows Variation of Dryness Fraction at Condenser inlet with Super heater Temperature

Figure 4 shows the variation of temperature in radial direction in fuel pellets, temperature distribution in fuel pellets decreases with increase in radial distance, temperature distribution in gas gap has sharp fall then the decrement of temperature distribution in cladding. Figure 5 shows the increment of efficiency with super heater temperature but heat input is increases with increase in super heater temperature shown in figure 6. Mass flow rate is decreases with increase in super heater temperature shown in figure 7. Figure 8-9 shows the increment of workout put and dryness fraction with super heater temperature.

4. CONCLUSION

As stream of water through the center is expanded, steam bubbles are all the more immediately eliminated from the center, the measure of fluid water in the center builds, neutron control expands, more neutrons are eased back to be consumed by the fuel, and reactor power increments. As stream of water through the center is diminished, steam voids stay longer in the center, the measure of fluid water in the center reductions, neutron control diminishes, less neutrons are eased back to be consumed by the fuel, and reactor power diminishes. In this work we concluded that Efficiency of power plant increases linearly as the superheating temperature increases. As the superheating temperature increases heat input also increases linearly. Mass flow rate across OFWH decreases as the temperature of super heating increases. Work output increases as the temperature of superheating increases. As the superheating temperature increases Dryness fraction of condenser also increases.

REFERENCES

[1] FRIGYES REISCH, High Pressure Boiling Water Reactor; KTH, Royal Institute of Technology, Nuclear Power Safety, Stockholm, Sweden.

[2] Hino, T., Miwa, J., Mitsuyasu, T., Ishii, Y., Ohtsuka, M., Moriya, K., & Greenspan, E. (2017). Core design and analysis of axially heterogeneous boiling water reactor for burning transuranium elements. Nuclear Science and Engineering, 187(3), 213-239.

[3] Wu, Z., Yang, W. S., Shi, S., & Ishii, M. (2016). A core design study for a small modular boiling water reactor with long-life core. Nuclear Technology, 193(3), 364-374.
[4] Mieloszyk, A. J., & Kazimi, M. S. (2015). Fuel performance analysis of a (ThU) O2-fueled, reduced moderation boiling water reactor. Nuclear Technology, 191(3), 268-281.
[5] Sukjai, Y., & Shirvan, K. (2021). Fuel performance analysis of reduced moderated boiling water reactor for transuranic waste incineration. Progress in Nuclear Energy, 137, 103738.
[6] Atkinson, S., Detkina, A., Litskevich, D., & Merk, B. (2021). A Comparison of Advanced Boiling Water Reactor Simulations between Serpent/CTF and Polaris/DYN3D: Steady State Operational Characteristics and Burnup Evolution. Energies, 14(4), 838.
[7] Hussain, T., Singh, A. K., Mittal, A., Verma, A., & Alam, Z. (2020). Performance Evaluation of Vapor Compression Refrigeration System by Varying Air Flow Rates in Air-Cooled and Evaporatively Cooled Condensers. International Journal of Energy for a Clean Environment, 21(1).
[8] Khan, K. F. A., Alam, Z., & Yahya, S. M. EXPERIMENTAL INVESTIGATION OF POTENTIALITY OF NANOFLUIDS IN ENHANCING THE PERFORMANCE OF HYBRID PVT SYSTEM.
[9] Hussain, T., Alam, Z., & Hafiz, A. EXPERIMENTAL INVESTIGATION OF CYLINDRICAL FOOD PRODUCT DURING FORCED CONVECTION COOLING.
[10] Alam, Z., Amir, M., Mansoor, M., & Ahmad, M. J. Cooling Load Estimation of a Multi-Storey Building: A Heat Transfer Approach.
[11] Hussain, T., Alam, Z., Ahamad, D., Anwar, M. S., Ahmed, S. I., & Hafiz, A. Dehumidification Analysis of Rotary Solid Desiccant Wheel & Packed Bed System.
[12] Michal Dudek and Marek Jaszczur, An analysis of the thermodynamics cycles with high temperature nuclear reactor for power generation and hydrogen co-production; AGH University of Science and Technology, Faculty of Energy and Fuels , Department of Research in Energy Engineering, Krakow, Poland.
[13] A.I.Oludare, M.N.Agu, P.O.Akusu and O.E.Omolara, Comparative Analysis of Water-Cooled Reactor Design Models and Gas –Cooled Reactor Design Models; Nigeria DefenceAcademy,Department of Physics, Kaduna; Nigeria Atomic Energy Commission,Abuja;Nigeria Atomic Energy Commission, Abuja; and Ahmadu Bell University (A.B.U.),Zaria ,Nigeria .
[14] Thermal Conductivity of Uranium Dioxide, INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1966
[15] AndreyIoilev, Maskhud Samigulin , Vasily Ustinenko; Advances in the Modeling of Cladding Heat Transfer and Critical Heat Flux in Boiling Water Reactor Fuel Assemblies,Russian Federal Nuclear Center (VNIIEF), Sarov ,Russia.