DUJASE Vol. 6 (2) 72-76, 2021 (July) *https://doi.org/10.3329/dujase.v6i2.59221*

Analysis of the Effect of Central Gap Size on the Temperature Profile and Fissile Content in the Annular Nuclear Fuel Rod

Farhana Islam Farha and Md. Hossain Sahadath*

Department of Nuclear Engineering, University of Dhaka, Dhaka-1000

**E-mail: hossain_ne@du.ac.bd*

Received on 30 May 2021, Accepted for publication on 13 September 2021

ABSTRACT

The perturbation in temperature profile and fissile content due to variation in the central gap of an isolated annular cylindrical fuel rod of a pressurized water reactor was studied by analytical calculation. Four different models namely UO_2+Zircaloy-4+He, UO_2+ Zr-1%Nb +He, MOX+Zircaloy-4+He, and MOX+ Zr-1%Nb +He were considered. The radial temperature profile was generated for different ratios (α) of outer to inner fuel radius. Lower fuel temperature was observed for small values of α and vice versa. The peak fuel temperature and temperature drop across the fuel pellets were calculated. Models of MOX fuel showed higher peak fuel temperature and large temperature drop than the models of UO_2 for the same fuel cladding. Zr-1%Nb cladding results in a slightly higher fuel temperature than Zircaloy-4 for the same fuel composition. The faster changes of these parameters with α were found for UO_2 than MOX fuel. The change in fissile loading with α was also studied and a sharp increase is observed if exceeds $\alpha = 2.50$.

Keywords: Annular Rod, Enrichment, Hole Size, MOX, Temperature, UO_2.

1. Introduction

The high specific power (power per unit fuel mass) and lower thermal conductivity of the oxide ceramic nuclear fuel result in an elevated centerline temperature in the solid cylindrical fuel rod at the operating condition of the nuclear reactor. Consequently, a large temperature gradient is developed across the fuel that favors the fuel pellet cracking alongside other radiation damage. This causes fuel failure which reduces fuel residence time at the reactor core and rods may be required to replace before the full burnup period [1]. The reactor transients, which are unavoidable events of reactor operation, may also result in high fuel temperature due to either under cooling or an increase in reactivity. Moreover, temperature escalation during accidental conditions even in design basis accidents (DBAs) may challenge the rod integrity which in turn promotes core damage [2]. To overcome these problems, the concept of annular fuel design and accident tolerant fuel has been introduced in nuclear reactor technology. In the annular fuel design, the cylindrical fuel rod, more specifically the nuclear fuel pellet comprises a central hole. Since there is no heat source in the central region, this design allows lower fuel temperature as compared to solid fuel rods of the same composition. However, higher enrichment is required to keep the power output the same [3]. There are two types of annular fuel design. In the outer cooled design, rods are cooled externally and the central hole is a vacuum. This design is modified to dual cooled annular rod where cooling is performed both internally and externally. The dual cooling design requires a greater amount of coolants and additional cost of inner cladding alongside higher enrichment due to less fuel. Therefore, outer cooled design is now used in some designs e.g. VVER-1000 [4] and dual cooled concept needs more research and development (R&D). In the present study, the outer cooling design has been considered. The temperature profile across the annular fuel rod depends on the chemical composition and physical properties of the nuclear fuel, cladding materials, gas used in the fuel cladding gap alongside the coolant flow characteristics, and last but not the least, the size of the central hole. The selection of nuclear fuel, cladding, and gap gas also considers the neutronic properties besides the thermal properties of these materials. This limits the choice to few materials which are now used in the form of the solid cylindrical fuel rod in most nuclear power plants (NPPs). If these materials are selected to design an annular rod, the central hole size plays a crucial role in the fuel temperature profile as other parameters remain the same [5]. Therefore, it is of utmost importance to study how fuel temperature changes with the size of the central annular hole. In addition to that, the change in fissile content caused by the change of fuel volume must also be studied to select the optimum annular gap size. Research works conducted previously regarding the annular fuel rod considered fixed central gaps to figure out the thermal and mechanical performance. Little works were carried out which deals with the dimensional variation of the central hole. This demands a study of the perturbation of fuel temperature and required fuel enrichment due to variation of the dimension of the annular gap. The present work was undertaken to study the impact of annular gap size on the fuel rod temperature and the fissile content considering different nuclear fuel and cladding.

Nomenclature

Symbol	Meaning (Unit)
H	Length of fuel rod ($\mathbf{m}$)
h	Thermal heat conductive coefficient of coolant ($\mathbf{W/m^2K}$)
h_g	Thermal heat conductive coefficient of Helium ($\mathbf{W/m^2K}$)
k	Thermal Conductivity of the coolant ($\mathbf{W/mK}$)

$\tilde{\mathbf{k}}_{\mathbf{f}}$	Average thermal heat conductivity of fuel $(\mathbf{W/mK})$
$\tilde{\mathbf{k}}_{\mathbf{c}}$	Average thermal heat conductivity of cladding $(\mathbf{W/mK})$
$\mathbf{q}'(\mathbf{r})$	Linear power $(\mathbf{W/m})$
$\mathbf{q}'''(\mathbf{r})$	Volumetric heat generation rate $(\mathbf{W/m^3})$
$\mathbf{r}$	Radial distance from the center of the fuel element $(\mathbf{m})$
$\mathbf{R_{ci}}$	Inner radius of cladding $(\mathbf{m})$
$\mathbf{R_{co}}$	Outer radius of cladding $(\mathbf{m})$
$\mathbf{R_{fi}}$	Inner radius of fuel $(\mathbf{m})$
$\mathbf{R_{fo}}$	Outer radius of fuel $(\mathbf{m})$
$\mathbf{R_g}$	Radius in gas gap $(\mathbf{m})$
$\mathbf{T_{ci}}$	Temperature at the inner surface of cladding material $(\mathbf{K})$
$\mathbf{T_{co}}$	Temperature at the outer surface of cladding material $(\mathbf{K})$
$\mathbf{T_m}$	Coolant bulk temperature $(\mathbf{K})$
$\mathbf{T_{max}}$	Maximum fuel centerline temperature $(\mathbf{K})$

The two types of nuclear fuel namely uranium dioxide (UO_2) and mixed oxide fuel (MOX) were considered. Zircalloy-4 and Zr-1%Nb alloy were used as fuel cladding while helium (He) was selected as gap gas. Zr-1%Nb is an alloy of zirconium with 1wt% of niobium. Niobium is added to increase the mechanical strength of the cladding. Four annular rods were modeled by combing these materials. A fuel temperature profile along the radial direction was generated. The peak fuel temperature and the temperature drop across the fuel pellet were determined. The change in active fuel volume and additional fissile fuel requirement were also studied.

2. Materials and Method

2.1 Geometry and properties of materials

An isolated externally cooled annular nuclear fuel rod is considered in this work. Table 1 contains the geometry specification and materials of the models. Fig. 1 shows the cross-section of a typical annular fuel rod. The fuel temperatures were determined for different values of fuel outer to inner radius ratio (α). This ratio was calculated by changing the inner radius of the fuel keeping the outer radius fixed. The outer radius cannot be changed as it changes the core geometry alongside the coolant flow channel. That is why the inner radius was taken as a variable. At first, the inner radius was taken 0.05334 cm to set the maximum value of 10. Then the different multiple of this inner radius was taken to calculate the other values of α. The burnup of nuclear fuel generates a large number of fission products alongside other major and minor actinides. Production of these elements changes the thermal conductivity of nuclear fuel (UO_2 & MOX) and is significantly different from the fresh unburned fuel. Since high burnup results in a reduction of thermal conductivity and high fuel temperature, data of thermal conductivity of fuel materials were considered for the highest burnup (5%).

Table 1: Specification of the models [6].

Reactor type	Pressurized Water Reactor (PWR)
Fuel material	UO_2 & MOX
Cladding material	Zircaloy-4 & Zr-1%Nb
Fuel rod (annular) outer radius, R_{fo}	0.5334 cm
Fuel rod (annular) inner radius, R_{fi}	Variable
Outer to inner radius ratio, $(\alpha = R_{fo}/R_{fi})$	Variable
Inner cladding radius, R_{ci}	0.5410 cm
Outer cladding radius, R_{co}	0.6019 cm
Thickness of cladding	0.0609 cm
Gap gas	Helium
Coolant	Water
UO_2+Zircaloy-4+He	Model 1
UO_2+Zr-1%Nb+He	Model 2
MOX+Zircaloy-4+He	Model 3
MOX+ Zr-1%Nb +He	Model 4

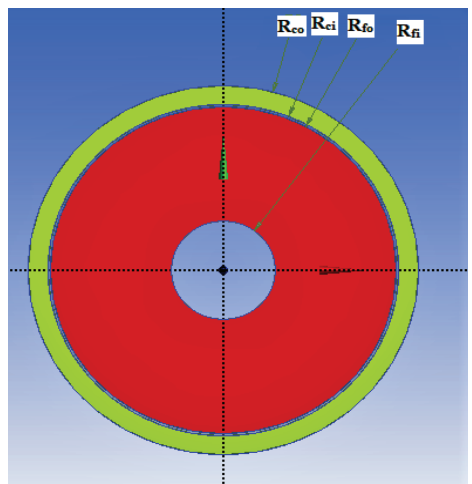

Fig. 1 Cross-sectional view of an annular fuel rod.

2.2 Analytical calculation

The peak fuel temperature, radial temperature distribution, and radial temperature drop across cladding and gas gap were calculated in the following way [6, 7].

Step 1: Fuel centerline temperature was found using equation

$$T_{max} = T_m + q'\left[\frac{1}{4\pi\widetilde{k_f}}\left[1-\frac{\ln\left(\frac{R_{fo}}{R_{fi}}\right)^2}{\left(\frac{R_{fo}}{R_{fi}}\right)^2-1}\right]+\frac{1}{2\pi R_g h_g}+\frac{1}{2\pi k_c}\ln\left(\frac{R_{co}}{R_{ci}}\right)+\frac{1}{2\pi R_{co} h}\right] \qquad (1)$$

Step 2: Radial temperature distribution throughout the fuel material was found using equation

$$T(r) = T_{max} - \frac{q'''}{4k}\left[r^2 - {R_{fi}}^2\right] + \frac{q''' {R_{fi}}^2}{2\widetilde{k_f}} \ln\left(\frac{r}{R_{fi}}\right) \quad \ldots\ldots\ldots\ldots\ldots\ldots\ldots (2)$$

Step 3: Radial temperature drop across Helium gas gap was found using

$$T_{co} = T_{ci} - \frac{1}{2\pi k_c} \ln\left(\frac{R_{co}}{R_{ci}}\right) \quad \ldots\ldots\ldots\ldots\ldots\ldots . (3)$$

Step 4: Radial temperature drop across cladding material was found using

$$T_{co} = T_{ci} - \frac{1}{2\pi k_c} \ln\left(\frac{R_{co}}{R_{ci}}\right) \quad \ldots\ldots\ldots\ldots\ldots\ldots . (4)$$

The above calculations were carried out in Microsoft Office Excel for all the models. The thermal conductivity of fuel, cladding, and gap gas play an important role in determining the fuel temperature distribution. This property is a function of temperature, material composition, level of fuel burnup, etc. In this study, the thermal conductivity of UO_2 & MOX fuel [8], Zircaloy-4 and Zr-1%Nb [9], and Helium [10] data were fetched from the literature.

3. Results and Discussion

3.1 Temperature Profile

Radial temperature distribution of annular fuel element with UO_2 fuel, Zircaloy-4 cladding, and Helium fuel-cladding gap is shown in Fig. 2. The temperature profile was observed with various sizes of annular holes in the centre of the fuel elements with a constant linear heat generation rate. To keep the linear heat generation rate constant, a higher level of fuel enrichment is required for larger holes. As the size of the annular gap increases, the maximum fuel centerline temperature decreases. Keeping the cost and risk of higher enrichments needed for larger annular holes in consideration, the size of the annular gap should be chosen, achieving a low centerline temperature, which is desired. Temperature drop across the gas gap is 321.5K and across cladding is 15.6K. The large temperature drop across the pellet/cladding gap is due to the lower thermal conductivity of the helium gas. The same is also true for other models and an identical value is observed. This is caused by the similar dimension of pellet/cladding gap and the same type of gap gas. The temperature drop across the cladding is lower due to its higher thermal conductivity alongside lower thickness. Change in cladding composition results in a slightly different temperature. Fig. 3 shows the same parameter for annular fuel element with UO_2 fuel, Zr-1%Nb cladding, and Helium fuel-cladding gap. Higher centerline temperatures are observed for all sizes of annular gap due to the use of Zr-1%Nb alloy as opposed to the use of Zircaloy-4 in Fig. 2. This is due to the lower thermal conductivity of Zr-1%Nb at the higher temperature [11]. Rests of the observations are similar in both cases. Temperature drop across the gas gap is 321.5K and across cladding is 17K. The radial temperature profile of the annular fuel element with MOX fuel, Zircaloy-4 cladding, and Helium fuel-cladding gap is shown in Fig. 4. Higher centerline temperatures are observed for all sizes of annular gap due to the use of MOX fuel as opposed to the use of UO_2 in Fig. 2 and Fig. 3. This can be explained by the lower burnup-dependent thermal conductivity of MOX fuel than UO_2 fuel [11]. Rests of the observations are similar to previous cases. Temperature drop across the gas gap is 321.5K and across cladding is 15.6K.

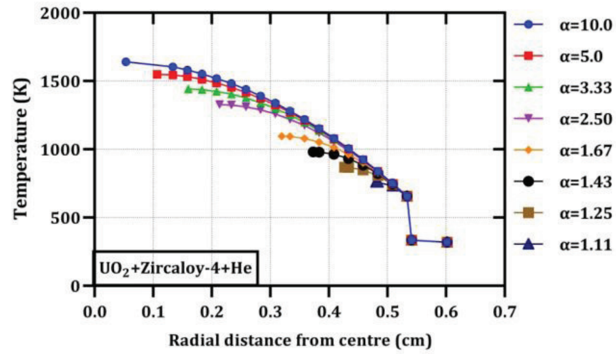

Fig. 2. Temperature distribution in annular fuel element with UO_2 fuel, He gas gap, and Zircaloy-4 cladding.

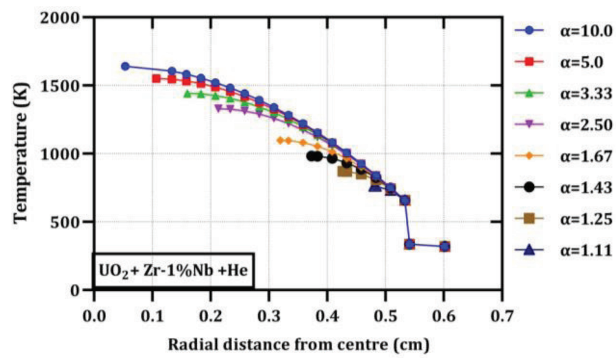

Fig. 3. Temperature distribution in annular fuel element with UO_2 fuel, He gas gap, and Zr-1%Nb cladding.

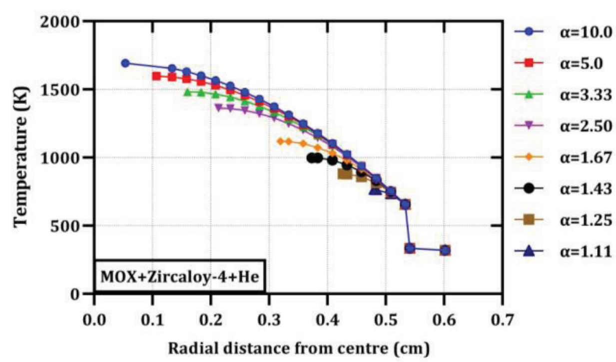

Fig. 4. Temperature distribution in annular fuel element with MOX fuel, He gas gap, and Zircaloy-4 cladding.

Fig. 5 depicts the temperature profile of Model 4 which comprises MOX fuel, Zr-1%Nb cladding, and Helium fuel-cladding gap. A similar variation with a slightly higher temperature than Fig. 4 is found. Temperature drop across the gas gap is 321.5K and across cladding is 17K. Table 2 and Table 3 list the peak fuel temperature and temperature drop respectively for each of the cases studied in the present work. These values are compared graphically in Fig. 6 and Fig.7. The highest centerline temperatures are observed in Model 4 for all sizes of annular gap due to the use of MOX fuel with ZrNb-1 cladding as opposed to the use of UO_2 with Zircaloy-4 or ZrNb-1 and MOX with Zircaloy-4 in other models. Similar results are found for the temperature drop. Fig. 8 is drawn dividing each PFT by the lowest PFT found at maximum gap size to visualize the comparatively faster change of peak fuel temperature of UO_2 fuel with the size of the annular gap.

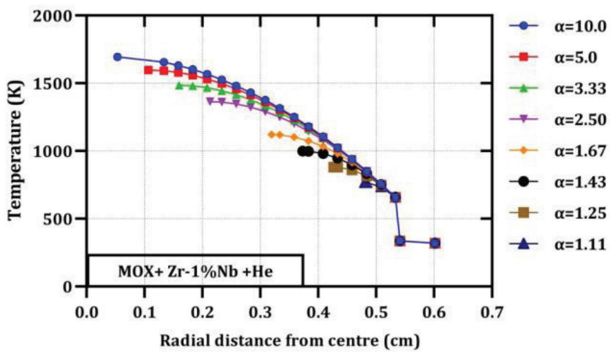

Fig. 5 Temperature distribution in annular fuel element with MOX fuel, He gas gap, and Zr-1%Nb cladding.

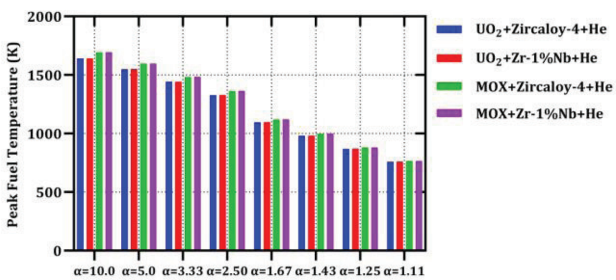

Fig. 6 Comparison of peak fuel temperature.

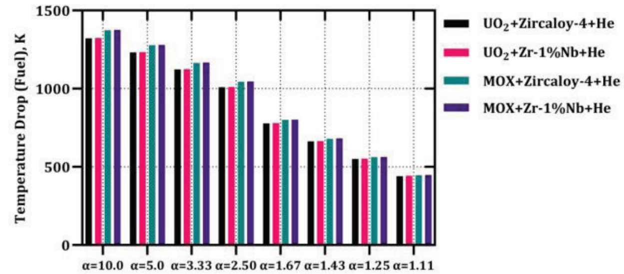

Fig. 7 Comparison of temperature drop across fuel element.

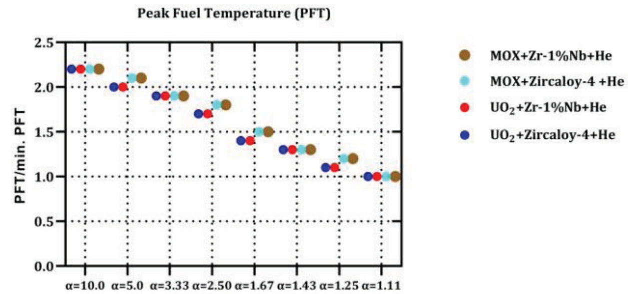

Fig. 8 Comparison of relative change in peak fuel temperature.

3.2 Fissile Content

Nuclides that undergo nuclear fission reaction with the neutron of any energy even zero are called fissile materials. These (e.g. U-235, Pu-239) are the main component of nuclear fuel. Annular gap size reduces fuel active volume which demands higher fuel enrichment i.e. fissile content for the same linear power rate (power produce per unit length of fuel rod). The total power produced from a single rod can be written as [6]

$$P = \sigma_f N \varphi E_R V \ldots\ldots\ldots\ldots\ldots\ldots\ldots\ldots (5)$$

Table 2: Maximum temperature of the fuel element.

Rod Composition	Peak Fuel Temperature (K)							
	α=10.0	α=5.0	α=3.33	α=2.50	α=1.67	α=1.43	α=1.25	α=1.11
UO_2+Zircaloy-4+He	1640.2	1549.7	1441.9	1327.7	1096.1	981.2	868.8	759.5
UO_2+Zr-1%Nb+He	1641.6	1551.1	1443.3	1329.1	1097.5	982.6	870.2	760.9
MOX+Zircaloy-4+He	1692.0	1596.8	1483.3	1363.1	1119.3	998.3	880.1	765.0
MOX+Zr-1%Nb+He	1693.4	1598.2	1484.7	1364.5	1120.7	999.7	881.5	766.4

Table 3: Temperature drop (ΔT) across the fuel element.

Rod Composition	Peak Fuel Temperature (K)							
	α=10.0	α=5.0	α=3.33	α=2.50	α=1.67	α=1.43	α=1.25	α=1.11
UO_2+Zircaloy-4+He	1321.8	1231.3	1123.5	1009.3	777.7	662.8	550.4	441.1
UO_2+Zr-1%Nb+He	1323.2	1232.7	1124.9	1010.7	779.1	664.2	551.8	442.5
MOX+Zircaloy-4+He	1373.6	1278.4	1164.9	1044.7	800.9	679.9	561.7	446.6
MOX+Zr-1%Nb+He	1375.0	1279.8	1166.3	1046.1	802.3	681.3	563.1	448.0

Here, $\sigma_f =$ microscopic fission cross-section, $N =$ atomic density of fissionable materials, $\varphi =$ neutron flux, $E_R =$ Recoverable fission energy, and $V =$ fuel volume. For convenience, we only considered fissile materials neglecting fertile nuclides. If annular gap size varies then only fissile density and fuel volume change in the eq. (5) for the same power output. Therefore we can rewrite the equation for two rods with different annular gaps and equating for the same rod power as

$$N_1 V_1 = N_2 V_2 \dots\dots\dots\dots\dots\dots\dots\dots (6)$$

Therefore, final fissile density can be found multiplying initial fissile density by the ratio of the fuel volume as

$$N_2 = N_1 (V_1/V_2). \dots\dots\dots\dots\dots\dots (7)$$

The fuel volume of the annular rod was calculated by the following equation.

$$V = \pi\left({R_{fo}}^2 - {R_{fi}}^2\right)H \dots\dots\dots\dots (8)$$

To assess the change in fuel volume and fissile content, relative values of these parameters were calculated taking values at $\alpha = 10.0$ as the initial value. Fig. 9 shows how these parameters change with annular gap size.

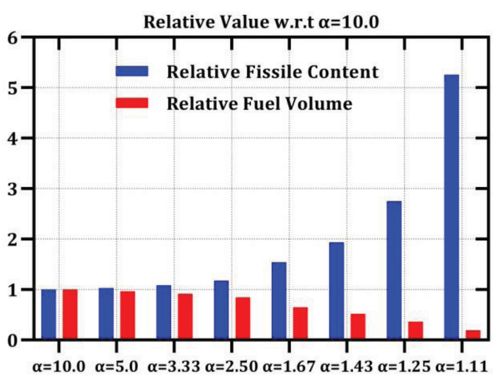

Fig. 9 Relative value of fuel volume and corresponding fissile content.

Fissile fuel loading increases rapidly at lower values of alpha i.e. for large central holes. Higher fissile loading means higher enrichment which adds extra cost to the fuel fabrication cost as well as generation cost. Therefore, there must be a balance between desired allowable fuel temperature and fuel enrichment.

4. Conclusion

The dimension of the central hole plays a crucial role in the thermal performance and required fuel enrichment of an annular nuclear fuel rod. The temperature profile of an isolated annular rod of a pressurized water reactor was assessed by changing the ratio (α) of the outer to the inner radius of the rod. The temperature distribution in the annular rod resembles the temperature profile of a solid nuclear fuel rod for all values of α. Lower temperature values were found for large annular gaps and vice versa. The UO_2 fuel showed a lower peak fuel temperature than MOX fuel for the same values of α and identical fuel cladding composition. This is due to the higher burnup-dependent thermal conductivity of UO_2 fuel than MOX fuel. On the other hand, the lower thermal conductivity of Zr-1%Nb results in a higher fuel temperature than Zircalloy-4 for the same fuel composition. Change in maximum fuel temperature with inner radius was found higher for UO_2 than MOX fuel. The same variation was observed for the temperature drop across the rod for both types of fuel. The relative fissile requirement increases sharply if α goes beyond 2.50. Therefore, it can be concluded that the fuel temperature can be reduced substantially by choosing lower values of α, however, there will be an economic penalty due to the higher cost of fuel enrichment. The annular gap size has a large impact on fuel enrichment and optimum value should be used to meet both thermal and fuel loading requirements alongside generation cost.

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