Calculation of fuel temperature profile for heavy water moderated natural uranium oxide fuel using two gas mixture conductance model for noble gas Helium and Xenon

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1. Introduction

Estimation of fuel temperature at all the operating conditions of the nuclear reactor is a very important issue as the safety of reactor operation hinges on fission heat extraction capability of the coolant. Error in heat and temperature estimation may have severe repercussions in terms of fuel overheating leading to clad failure, fuel centre achieving melting point thereby releasing fission gas trapped in the pellets. This fission gas is highly radioactive and can lead to spread of contamination in the coolant tubes. During reactor transients, the fuel pellet may release stored heat, it may lead to spikes in fuel temperature for all operating conditions of the reactor is a very important as reactor operation depends upon flow obstruction, or there is a local power excursion, the fuel rod contains a stack of natural uranium oxide pellets 12.1 mm in diameter within a Zircalloy-4 sheath of wall thickness 0.4 mm. The fuel stack length is 49.5 cm. Neutronically, TAPS-3 & 4 twin units operating reactor at power, the UO2 has a high centre temperature with respect to its surface temperature.

For comparison and validation of our model, we have used operating conditions of Pressurized Heavy Water Reactors operating at Tarapur Atomic Power Station (TAPS-3&4), India. In this reactor the fuel bundle consists of an assembly of 37 cylindrical fuel elements in eighteen, twelve, six, and one fuel rod on circles. Each fuel rod contains a stack of natural uranium oxide pellets 12.1 mm in diameter within a Zircalloy-4 sheath of wall thickness 0.4 mm. The fuel stack length is 49.5 cm. Neutronically, TAPS-3&4 twin units are large loosely coupled PHWRs and accurate estimation of fuel temperature for all operating conditions of the reactor is a very important as reactor operation depends upon fission heat extraction capability of the coolant. The fission gas is highly radioactive and can lead to spread of radioactive contamination in the coolant tubes. During reactor transients when coolant flow is unsteady due to flow obstruction, or there is a local power excursion, the fuel pellet may release stored heat, it may lead to spikes in fuel conductivity of UO2 is dependent upon temperature. In an operating reactor at power, the UO2 has a high centre temperature with respect to its surface temperature.

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temperature. Perturbation due to reactivity device movement may also lead to local change in fission power and change the fuel temperature in a local region. As coolant flow is steady, fuel temperature may be higher than that calculated by 3-D neutron diffusion equation solver code such as TRIVENI developed by a group led by Srinivasan and Purandare [23]. TRIVENI is a validated code available with IAEA software database [9,10]. In Indian PHWR fuel design the Zircalloy fuel sheath thickness is approximately 0.40 mm and the sheath may collapse on the fuel pellet due to coolant pressure. Hence pellet temperature directly affects pellet-clad interaction phenomenon.

As discussed earlier, change in conductivity due to fission gas accumulation in the pellet-sheath gap is the main thrust of this paper, and this effect is observable during refuelling when out of the thirteen irradiated fuel bundles in a channel, eight irradiated fuel bundles are replaced by fresh natural uranium oxide bundles. Higher fission rate and higher flux due to the fresh bundles lead to local increase in temperature of the older five bundles in the same channel. Increase in temperatures of the old bundles causes higher fission gas release in the pellet to fuel-sheath gap and hence change in temperature of the old bundles. In the work presented in this paper, a computer code Fuel Characteristics Calculator (FCCAL) based on binary gas model is developed to analyze this phenomenon. Results obtained from FCCAL show that this phenomenon is captured in the code where although the fuel bundles are producing same power, but they have different temperature due to different irradiation. This is because different irradiation translates to different burnup and hence different amount of fission gas in the fuel. Moreover, fuel temperature calculation gives us an estimate of the heat load of fuel residing in the core. When operating in a reactor at power, the UO₂ has a high centre temperature with respect to its surface temperature.

Under normal operating conditions, the fission gas generation rate is about 0.3 atoms per fission event [13,30]. It is generally agreed that below 600°C – 800°C, the rate of fission gas release is controlled by recoil and knock out [25]. In the region between 800°C and 1800°C, fission gas release rates are higher and markedly temperature dependent [18]. Accumulation of fission gas severely affects the heat carrying capacity of the fuel rod assembly in steady state as well as reactor transient operating conditions, [1,11]. A widely accepted and benchmarked code GRASS [17] provides a detailed model for fission gas diffusion within the ceramic fuel and its migration to grain boundaries. GRASS gives a phenomenological model for estimation of fission gas accumulating in the pellet to fuel sheath gap interspaces. However, runtime of the code GRASS is high and hence various alternate models [15,28] have been developed for quick calculation of fission gas accumulating in the gap e.g., BUBL-1 [27]. Work done by several researchers to develop models for estimation of fuel temperature has yielded many reliable and useable models [19,21]. A wide range of dedicated software developed for the estimation of fuel temperature profile bear testimony to the importance of this issue [29].

Presently, computer code FUDA [2,5,26] is being used for Indian PHWRs for study of fuel behaviour. FUDA has been developed for fuel element operation history and fuel element thermo-mechanical analysis of Indian PHWRs. It is used for modelling PHWR fuel bundle radial temperature profile across fuel and sheath, fission gas release, internal gas pressure, sheath stress and strains during the life of fuel bundle. Gap gas conductance model of Lassmann and Pazdera [12] is used in FUDA. Our work improves upon the existing code by carrying out an explicit calculation for changes in gap gas conductivity due to binary mixture of Helium and Xenon. As heat generated in the pellet is transferred across the fuel pencil to the coolant, the heat transfer discontinuities and the corresponding heat transfer phenomena occur according to the following steps.

- Heat transfer from the “meat” of the uranium oxide pellet to the pellet surface,
- Heat transfer from pellet surface to the Zircalloy sheath across the gap and,
- Heat transfer from collapsible Zircalloy cladding surface to the coolant.

For gap gas conductivity calculation two case were considered. In first case, analysis was carried out if the gas is a binary mixture of He–Xe and in the second case, analysis was done for the gas binary mixture of He–Kr. In this paper, parametric study has been done for the fuel bundle producing nearly equal powers, but having different burn-ups and hence different fission gas inventories inside the gap. We have calculated fuel temperature for various operating powers of the reactor using FCCAL. The result obtained from this code has been used in 3-D neutron diffusion calculation code.

The paper is organised along the following lines; Section 2 presents a discussion of the calculation methodology and is further subdivided into subsections detailing the heat transfer calculation across the discontinuities, the details of implementation of the model into code FCCAL is given in Section 3. The obtained results are discussed in Section 4. Section 5 presents conclusion and further scope of the work.

2. Calculation methodology

Heat transfer across the fuel pencil occurs via three intervening media.

(a) Heat transfer across the fuel pellet. It is estimated by heat transfer coefficient of the natural uranium oxide fuel pellet - \( h_P \)

(b) Heat transfer across the gap and across the Zircalloy sheath - \( h_{Gg} + h_{Gs} + h_S \). It is estimated by heat transfer coefficient of the pellet surface to the Zircalloy outer surface in the following sub-steps.

- Heat transfer coefficient of the gap between the fuel pellet and the Zircalloy-4 sheath - \( h_{Gg} \)
- Heat transfer coefficient of the solid-solid contact points of between the pellet and the Zircalloy-4 sheath - \( h_{Gs} \)
- Heat transfer coefficient of the Zircalloy-4 sheath - \( h_S \)

(c) Heat transfer from the pencil surface to the coolant. It is estimated by heat transfer coefficient of the coolant film near the fuel surface - \( h_{CF} \).

The total heat transfer coefficient \( h_{TF} \) for a fuel pencil is the sum of terms ‘a’,‘b’ and ‘c’ mentioned above i.e., Term (a) + Term (b) + Term (c) = \( h_{TF} = h_P + h_{Gg} + h_{Gs} + h_S + h_{CF} \).

2.1. Fuel pellet conductivity and temperature calculation

PHWR fuel pellet is made of UO₂ ceramic containing 0.7% U-235. It has poor heat conduction properties as compared to uranium carbide, or metallic uranium. The heat transfer coefficient of UO₂ is dependent upon temperature as well as fission product accumulating in the pellet. Moreover, as the fuel undergoes irradiation, cracks develop in the pellet and that also changes the heat transfer coefficient. Calculation of fuel pellet heat transfer coefficient \( h_P \) has been done using correlation from MATPRO [18] fuel analysis package. MATPRO is a computer library of 39 subcodes dealing with uranium dioxide and mixed uranium-plutonium dioxide fuel, Zircalloy cladding, gas mixture, and LWR fuel rod material properties.
Each property is programmed in MATPRO as a separate unit, either as a function or as a subroutine, so that individual correlations may be altered without changes being made elsewhere in the program. Fuel thermal conductivity calculations are provided in the module Fuel Thermal Conductivity - FTHCON of MATPRO package. A semi empirical correlation employing an analytical expression suggested by theory with constants determined by comparison with data pooled from ten sources is used for fuel thermal conductivity calculation. The thermal conductivity of un-irradiated UO$_2$ is well documented, especially in the temperature range below 1400°C. The correlations used in MATPRO to fit the data are:

\[ 0 \leq T \leq 1650^\circ C \]

\[ h_p = \eta \frac{B_1}{B_2 + T} + B_3 e^{(B_4 T)} \]  

(1)

where, \( h_p \) = W/cm$^0$C and \( T \) is in $^0$C

\[ 1650 \leq T \leq 2940^\circ C \]

\[ h_p = \eta \left[ B_5 + B_3 e^{(B_4 T)} \right] \]

(2)

where, \( \eta = \left[ \frac{1 - e^{(1 - \frac{1}{B_3})}}{1 - \eta (1 - 0.95)} \right] \) is porosity correction factor.

The constants are.

The integral of UO$_2$ thermal conductivity between $0^\circ$C and the melting point 2840°C is analytically determined in MATPRO. Assuming that the electronic contribution $B_3 e^{(B_4 T)}$ has the value of $2 \times 10^{-3}$Wcm$^{-1}$K$^{-1}$ at 1500°C, a least squares value of 97Wcm$^{-1}$ is obtained for the integral of $h_p$ from $0^\circ$C to the melting point. Plot generated from the available data is shown in Fig. 1.

Data points were fit to an equation including a temperature-dependent, modified Loeb [14] porosity correction. The cause of data scatter in Fig. 1 includes pellet cracking, relocation, irradiation, and differences in oxygen to metal ratios. In fuel rods having a gap width greater than about one percent of the diameter, cracking, bulk relocation of the oxide may occur and result in apparent conductivities different from that shown here. Stoichiometry also affects the thermal conductivity. Most data indicate an enhancement in $h_p$ [3,7] for hypostoichiometric samples and degradation in $h_p$ [20] for hyperstoichiometric samples. For UO$_2$ fuel the constants which yield the smallest standard deviation with respect to the data are 40.4 and 464 respectively. To obtain agreement between the high $T > 1650^\circ$C and low $T < 1650^\circ$C temperature portions of the curve, $B_3$ is taken as 0.0191Wcm$^{-1}$K$^{-1}$. Value of $B_1$ and $B_4$ affect the curve above 500°C at the low temperature end of the curve shown in Fig. 2.

Hence values for $B_3$ and $B_4$ are chosen such that they do not affect good fit for $T < 1650^\circ$C and 1650°C < $T < 2840^\circ$C. This is achieved by integrating Equation (1) and Equation (2) from $0^\circ$C to melting point $T_{mel}$ by assuming a value for the integral and numerically solving the resulting equation shown below.

\[ \frac{T_{mel}}{C_k} \quad \int \frac{T_{mel}}{C_k} \quad \int \frac{T_{mel}}{C_k} \quad \int \frac{T_{mel}}{C_k} \quad \int \frac{T_{mel}}{C_k} \]

(3)

\[ B_1 \left[ e^{(B_4 \times 2840)} - 1 \right] = \varphi - 83.99 \]

(4)

In addition to Equation (3) and Equation (4) for the two unknowns, we need another condition to obtain $B_3$ and $B_4$. For $T < 1650^\circ$C the data shows no significant deviation from the hyperbolic part of Equation (1) i.e. $\frac{B_3}{B_4}$. Since the data is accurate to ±5% at 1500°C the electronic contribution $k_e = B_3 e^{(B_4 T)}$ to the thermal conductivity is ≤ 0.002Wcm$^{-1}$K$^{-1}$. If $k_e < 0.002Wcm^{-1}$K$^{-1}$ at 1500°C then Fig. 2 shows a good fit for $T < 1650^\circ$C cannot be obtained. This is not desirable hence, $k_e = 0.002Wcm^{-1}$K$^{-1}$ is assumed and another equation for $B_3$ and $B_4$ is obtained as follows

\[ B_3 e^{(B_4 \times 1500)} = 0.002Wcm^{-1}K^{-1} \]

(5)

It should be noted that rather than taking any other equation form e.g., polynomial, the exponential term in Equation (1) and Equation (2) is used because it has the same general form as the

![Fig. 1. Comparison of measured and predicted values of the thermal conductivity of UO2. The result is for materials corrected to 95% Theoretical Density and standard deviation of data from theoretical curve (MATPRO Version – 10, Figure A-2.1).](image-url)
theoretical equation for the temperature dependence of the density of conduction band electrons, and the contribution to $h_p$ from these electrons should become important at high temperatures. Using Equation (1), Equation (2) and Equation (5) for different values of $\int k dT$ we find that the value of $\varphi$ giving the smallest standard deviation is $\int^{T}_C k dT = 97 \text{Wcm}^{-1}$. The values of $B_2$ and $B_4$ resulting from different assumed values of conductivity integral of the electronic contribution are shown in Table 1.

In order to widen the applicability of our model we have taken another model for pellet heat conductivity based on Westinghouse formula [24] for uranium oxide pellet thermal conductivity. The user has an option to use whichever model he/she finds pertinent for calculations. In the Westinghouse model for fuel pellet conductivity $h_p$ is given by the expression

$$h_p = \frac{1}{11.8 + 0.02387} + 8.775 \times 10^{-13} T^3$$

$$h_p = \frac{W}{cm^0 C}, T = 0 C$$

The results obtained by FCCAL were compared with TRIVENI and we found that MATPRO model gives a closely matching result as shown in Fig. 3 and hence it is used in this work.

2.2. Equivalent conductivity and temperature drop across the gap between fuel pellet and zircalloy sheath

Heat transfer coefficient $h_{GR}$ due to fission gas accumulating in the gap between the fuel pellet and the Zircalloy-4 sheath is a function of the fission gas diffusing from the pellet towards the gap. For fresh fuel the conductivity of the gap is a function of Helium thermal conductivity but the fission gas accumulating in the gap changes gap conductivity. Helium is present in the gap due to Helium filled in the pencil during manufacturing. Change in composition of gap gas is a function of the fission gas accumulating in the gap and it has been estimated using the conventional industry standard for estimation of fraction of Xe and Kr diffusing from the pellet to the gap [16] as shown in Table 2.

The yield of Xenon and Krypton is appreciable in the fission process and as they diffuse towards the gap, they affect the gap gas conductivity with fuel burn up. Thermal conductivity of the gas mixture consisting of He and fission gas accumulating in the pellet-fuel sheath gap is calculated using n-component gas mixture thermal conductivity model of Saksena and Saxena [22]. Equation (6) gives the values of $\lambda_{mix}$ and $\lambda_i$ are the thermal conductivities of the mixture gas and of the individual gases respectively, $X_i$ and $X_j$ are the mole fractions of the component gases, $\varphi_{ij}$ is constant.

For the binary gas mixture consisting of He–Kr or He–Xe the equation may be modified and written as

$$\lambda_{mix} = \lambda_1 \frac{1}{1 + \varphi_{12} X_i} + \lambda_2 \frac{1}{1 + \varphi_{21} X_j}$$

Where $\lambda_{mix}$, $\lambda_1$ and $\lambda_2$ are the thermal conductivities of the binary mixture and of the two gases respectively, $X_i$ and $X_j$ are the mole fractions of the two components in the mixture, and $\varphi_{12} = 0.3849$ and $\varphi_{21} = 3.4284$ for He–Kr binary mixture. $\varphi_{12} = 0.4909$ and $\varphi_{21} = 2.7863$ for He–Xe binary mixture.$X_1$ and $X_2$ is dictated by fuel burnup and temperature of fuel as they are function of fission gas accumulating in the fuel pencil-clad gap. The subscript 1 refers to the heavier gas in the binary mixture; hence 1 refers to Xe in He–Xe binary mixture and Kr in He–Kr binary mixture. Thermal conductivity of the binary mixture of (a) He and Xe and (b) He and Kr is calculated using gas mixture conductance model and the temperature profile is shown in Fig. 4.

Table 1

<table>
<thead>
<tr>
<th>$\int k dT$</th>
<th>$B_3$</th>
<th>$B_4$</th>
<th>$k_e - B_4 \varphi^{\beta} T$ at $1500^\circ C$ (Wcm$^{-1}$K$^{-1}$)</th>
</tr>
</thead>
<tbody>
<tr>
<td>93</td>
<td>2.535 $\times 10^{-4}$</td>
<td>1.377 $\times 10^{-3}$</td>
<td>0.002</td>
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<tr>
<td>94</td>
<td>2.024 $\times 10^{-4}$</td>
<td>1.627 $\times 10^{-3}$</td>
<td>0.002</td>
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<tr>
<td>95</td>
<td>1.688 $\times 10^{-4}$</td>
<td>1.656 $\times 10^{-3}$</td>
<td>0.002</td>
</tr>
<tr>
<td>96</td>
<td>1.412 $\times 10^{-4}$</td>
<td>1.767 $\times 10^{-3}$</td>
<td>0.002</td>
</tr>
<tr>
<td>97</td>
<td>2.116 $\times 10^{-5}$</td>
<td>2.555 $\times 10^{-3}$</td>
<td>0.002</td>
</tr>
<tr>
<td>98</td>
<td>1.216 $\times 10^{-4}$</td>
<td>1.867 $\times 10^{-3}$</td>
<td>0.002</td>
</tr>
<tr>
<td>99</td>
<td>1.064 $\times 10^{-4}$</td>
<td>1.956 $\times 10^{-3}$</td>
<td>0.002</td>
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</table>
In Fig. 4 the difference in fuel temperature is because of the composition of the gas mixture. It is expected as the yield of Kr is small and hence He dominates in equivalent thermal conductivity of the gap if a binary mixture of He-Kr is considered. Although the binary gas mixture of Xe–He is considered but the effect of fission gas accumulation with burn up for fuel producing nearly equal power is visible in Fig. 5 and it brings out the difference in fuel centreline temperatures due to different amount of fission gas accumulated in the gap. We can see in Fig. 5 that although the fuel bundles may be producing nearly equal power, since different amount of fission gas has accumulated in the gap it causes higher fuel centreline temperature for higher burn up fuel due to reduced conductivity caused by poor heat conduction properties of Xenon accumulating in the gap. Average gap temperature is about 8°C higher than the coolant temperature as calculated in the code and matches with the assertion in the FRAPTRAN-2.0 manual.

### Table 2

<table>
<thead>
<tr>
<th>LowerTemperatureLimit</th>
<th>HigherTemperatureLimit</th>
<th>Fraction</th>
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<tbody>
<tr>
<td>&lt; 1400°C</td>
<td>1500°C</td>
<td>0.05</td>
</tr>
<tr>
<td>1400°C</td>
<td>1500°C</td>
<td>0.10</td>
</tr>
<tr>
<td>1500°C</td>
<td>1600°C</td>
<td>0.10</td>
</tr>
<tr>
<td>1600°C</td>
<td>1700°C</td>
<td>0.10</td>
</tr>
<tr>
<td>1700°C</td>
<td>1800°C</td>
<td>0.10</td>
</tr>
<tr>
<td>1800°C</td>
<td>&gt; 2000°C</td>
<td>0.98</td>
</tr>
</tbody>
</table>

### 2.3 Zircalloy sheath conductivity and temperature estimation

Zircalloy sheath heat transfer coefficient $h_s$ is calculated based on Zircalloy conductivity expression from MATPRO using the expression

$$j_{\text{mix}} = \frac{\sum x_i l_i}{\sum x_i (1 + \rho_0)^{1/3} l_i}$$

(6)

**Fig. 3.** Error in average coolant temperature estimation using FCCAL with respect to TRIVENI calculation for 100% reactor power. Error = \( \frac{\text{TRIVENI} - \text{FCCAL}}{\text{TRIVENI}} \) × 100

**Fig. 4.** Temperature profile of fuel pencil with gap gas consisting of He–Xe binary and/or He–Kr binary for bundle at string position 6 for reactor channel E–06 for 100% reactor power.

In Fig. 4 the difference in fuel temperature is because of the composition of the gas mixture. It is expected as the yield of Kr is small and hence He dominates in equivalent thermal conductivity of the gap if a binary mixture of He–Kr is considered. Although the binary gas mixture of Xe–He is considered but the effect of fission gas accumulation with burn up for fuel producing nearly equal power is visible in Fig. 5 and it brings out the difference in fuel centreline temperatures due to different amount of fission gas accumulated in the gap. We can see in Fig. 5 that although the fuel bundles may be producing nearly equal power, since different amount of fission gas has accumulated in the gap it causes higher fuel centreline temperature for higher burn up fuel due to reduced conductivity caused by poor heat conduction properties of Xenon accumulating in the gap. Average gap temperature is about 8–10°C higher than the coolant temperature as calculated in the code and matches with the assertion in the FRAPTRAN-2.0 manual.

**Fig. 5.** Temperature profile of fuel pencil with gap gas consisting of He–Xe binary for bundle at string position 6 for reactor channel E–06 and bundle at string position 4 for reactor channel F-11 for 100% reactor power.
Fig. 6. Average coolant temperature calculated from the code FCCAL for 100% reactor power.

Fig. 7. Average coolant temperature calculated from the code TRIVENI for 100% reactor power.
hs = \left[ 7.51 + 2.09 \times 10^{-2} - 1.45 \times 10^{-5} T^2 + 7.67 \times 10^{-9} T^3 \right] \quad (7)

where, $h_s$ is in $W/mK$ and $T$ is in $K$.

2.4. Estimation of heat transfer and temperature drop across coolant

Coolant heat transfer coefficient $h_{Cf}$ is calculated by Colburn [4] for heat transfer given by NusseltNo.

\[ N_{D} = \left( \frac{h_{D}}{k} \right) = \frac{0.023 R e^{0.8} P r^{-0.2}}{ \text{Re and Pr have their usual meanings.} } \]

The average coolant temperature obtained by using Colburn correlation is closely matching with 3-D neutron diffusion equation solver TRIVENI as seen in Fig. 6. Coolant heat conductivity $h_{Cf}$ calculated by Colburn correlation is used in this paper.

3. FCCAL code methodology

FCCAL is written in FORTRAN language. It implements the model discussed in this paper and can be used to calculate fuel centreline temperature, fuel pencil surface temperature and average coolant temperature in steady state (as well as transient conditions) as a function of coolant flow rate ($Kg/s$), bundle power ($Kw$), fuel burnup, average channel power ($Kw$) and actual coolant temperature.

Fig. 8. Fuel centreline temperature calculated using code FCCAL for 100% reactor power.

Fig. 9. Fuel pencil surface temperature calculated using code FCCAL for 100% reactor power.
Coolant outlet temperature of 392 channels are calculated in FCCAL (Fig. 10) and in TRIVENI (Fig. 11) for 50% full power. Study of Figs. 3, 6 and 7 shows a good match in the results. For the start of iteration in FCCAL a guess value of fuel centreline temperature at 2000°C is assumed taking into account the fact that the fuel centreline temperature cannot be less than the coolant temperature when the reactor is operating at a steady state. This is because power generated in the fuel is taken away by the coolant and transferred to the steam generator and since fuel is the source of heat, it is necessarily at a higher temperature as compared to the coolant. Equivalent thermal conductivity of fuel pellet, gas gap, and Zircalloy sheath are calculated in the code (FCCAL). Starting with the guess value of fuel centreline temperature, the temperature drop across the fuel pellet, gas gap, Zircalloy sheath and the coolant temperature are calculated using the equivalent conductivity across each discontinuity. The result is then compared with the coolant temperature calculated in TRIVENI for the corresponding channel. If the error in estimation is not within ±20°C, a revised estimation of fuel centreline temperature is made and the iteration is repeated until the error in estimation is within ±20°C. When the coolant outlet temperature of 392 channels is calculated in TRIVENI for 100% full power is shown in Fig. 7.

Fig. 10. Average coolant temperature calculated using code FCCAL for 50% reactor power.

Fig. 11. Average coolant temperature calculated from the code TRIVENI for 50% reactor power.
corresponding error is within limits, the calculated fuel centreline temperature, fuel pencil surface temperature, and average coolant temperature calculated by the code is taken as correct. In steady state reactor operation, the coolant temperature is a reliable indicator of the fuel temperature. The fuel temperature and fuel heat content thus obtained is then used as input to estimate the fuel temperature profile in transients, when the steady state heat transfer between the fuel and the coolant is no longer the case. In such cases, the specific heat capacity of UO₂ and Zircalloy also acquire significance and cannot be neglected in estimation of the total heat output and heat removal requirement of the fuel assembly. Hence in that case the temperature profile of the fuel pencil along with the heat capacity of UO₂ and Zircalloy gives the total heat of the assembly.

4. Results and discussion

Channel power, coolant flow and channel coolant output temperature from channel temperature monitoring system for 392 channels of 540MWe Indian PHWR at Tarapur is taken as input in the code. Fuel centreline temperature, fuel pencil surface temperature, and average coolant temperature calculated in FCCAL is shown in Fig. 12.

Fig. 12. Fuel pencil surface temperature calculated using code FCCAL for 50% reactor power.

Fig. 13. Fuel centreline temperature calculated using code FCCAL for 50% reactor power.
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Appendix A. Supplementary data

Supplementary data to this article can be found online at https://doi.org/10.1016/j.net.2020.05.024.

Nomenclature

<table>
<thead>
<tr>
<th>Abbreviation</th>
<th>Meaning</th>
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<tr>
<td>FCCAL</td>
<td>Fuel Characteristics Calculator</td>
</tr>
<tr>
<td>FUDA</td>
<td>Fuel Design Analysis</td>
</tr>
<tr>
<td>( h_p )</td>
<td>Heat transfer coefficient of fuel pellet (W/m²K)</td>
</tr>
<tr>
<td>( h_{gg} )</td>
<td>Heat transfer coefficient of gap gas (W/m²K)</td>
</tr>
<tr>
<td>( h_{ss} )</td>
<td>Heat transfer coefficient of solid-solid contact (W/m²K)</td>
</tr>
<tr>
<td>( h_t )</td>
<td>Heat transfer coefficient of zircalloy sheath (W/m²K)</td>
</tr>
<tr>
<td>( h_f )</td>
<td>Heat transfer coefficient of coolant film across the fuel pencil surface (W/m²K)</td>
</tr>
<tr>
<td>LWR</td>
<td>Light Water Reactor</td>
</tr>
<tr>
<td>MWe</td>
<td>Mega Watt Electrical</td>
</tr>
<tr>
<td>MWth</td>
<td>Mega Watt Thermal</td>
</tr>
<tr>
<td>PHWR</td>
<td>Pressurized Heavy Water Reactor</td>
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<tr>
<td>TAPS-3&amp;4</td>
<td>Tarapur Atomic Power Station Units-3&amp;4</td>
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Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.


