THERMAL BEHAVIOR ANALYSIS OF PWR FUEL DURING RIA AT VARIOUS FUEL BURNUPS USING MODIFIED THEATRe CODE

by

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The fuel irradiation and burnup causes geometrical and dimensional changes in the fuel rod which affects its thermal resistance and ultimately affects the fuel rod behavior during steady-state and transient conditions. The consistent analysis of fuel rod thermal performance is essential for precise evaluation of reactor safety in operational transients and accidents. In this work, analysis of PWR fuel rod thermal performance is carried out under steady-state and transient conditions at different fuel burnups. The analysis is performed by using thermal hydraulic code, THEATRe. The code is modified by adding burnup dependent fuel rod behavior models. The original code uses as-fabricated fuel rod dimensions during steady-state and transient conditions which can be modified to perform more consistent reactor safety analysis. AP1000 reactor is considered as a reference reactor for this analysis. The effect of burnup on steady-state fuel rod parameters has been investigated. For transient analysis, hypothetical reactivity initiated accident was simulated by considering a triangular power pulse of variable pulse height (relative to the full power reactor operating conditions) and pulse width at different fuel burnups which corresponds to fresh fuel, low and medium burnup fuels. The effect of power pulse height, pulse width and fuel burnup on fuel rod temperatures has been investigated. The results of reactivity initiated accident analysis show that the fuel failure mechanisms are different for fresh fuel and fuel at different burnup levels. The fuel failure in fresh fuel is expected due to fuel melting as fuel temperature increases with increase in pulse energy (pulse height). However, at relatively higher burnups, the fuel failure is expected due to cladding failure caused by strong pellet clad mechanical interaction, where, the contact pressure increases beyond the cladding yield strength.

Key words: reactivity initiated accidents, fuel rod thermal behavior, cladding deformation, AP1000, THEATRe code

INTRODUCTION

In 1960s to early 1970s, the safety criteria of nuclear fuel were established [1] and revised afterwards according to the progress and developments in the nuclear industry. The reactivity initiated accident (RIA) is a design based accident and must be analyzed according to the established safety criteria in order to protect the core from severe accident. The primary initiating cause of RIA is considered as the sudden control rod ejection from the core which result in rapid increase in core power and its temperature.

The fuel rod behavior under RIA can be investigated through experiments and through computer simulations. Although, the significance of experimental analysis for assessment of safety related issues cannot be neglected, however, the safety analysis using computer simulation is an economical way as compared with the experimental analysis. Several computer codes have been developed to simulate and understand the fuel rod behavior. Moreover, many computer codes have been modified to predict the fuel rod behavior for more realistic best estimate analysis of design base accidents. Comprehensive modeling of RIA require the simultaneous solution of the equations representing neutron transport, heat transfer within fuel rod (from fuel to cladding and to the coolant), thermomechanical behavior of fuel rod and the coolant thermal hydraulics [2]. Generally, computer codes are developed for a specific analysis such as thermal hydraulic, reactor kinetics, fuel rod behavior *etc*. These codes normally focus on specific analysis for which they are developed and the effect of other disciplines may be ignored or approximated. Therefore, it

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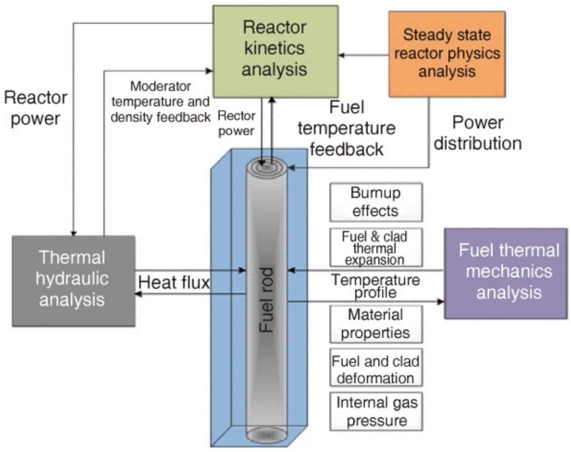

Figure 1. Relationship between important nuclear reactor analysis tools [4]

may neither provide consistent representation of the entire phenomena nor achieve realistic and consistent results. The accuracy of computer codes designed for reactor safety analysis is directly related to the detailed and precise mathematical representation of complex phenomena and processes associated with nuclear power plant. The consistent computation method of multi-physics systems have been started globally to overcome this issue [3, 4]. The block diagram representing relationship between important nuclear reactor analysis tools is given in fig. 1.

Several fuel rod behavior analysis computer codes are available for steady state and transient analysis. Researchers have made many efforts to couple the thermal hydraulic and fuel rod behavior simulation tools in order to achieve realistic and consistent results. A brief overview of some of the codes is given in the following text.

The coupled codes RELAP5/SCDAP is used for best estimate thermal hydraulic analysis of nuclear power plants in case of severe accident. The code is developed by Idaho National Engineering Laboratory (INEL). The code takes into account some simplified fuel rod behavior models including cladding oxidation and ballooning, gas-gap pressure, release of fission gas, deformation due to creep, melting of fuel rod, fuel disintegration, fuel rod rupture, *etc*. SCDAP/RELAP5 code utilizes MATPRO material properties [4, 5].

For steady-state and transient fuel rod behavior analysis, FRAPCON and FRPTRAN codes are one of the latest and advanced computer codes, respectively. These codes are developed by Pacific Northwest National Laboratory for the analysis of LWR oxide fuel rods [6, 7].

SCANAIR code is developed by L'Institute de Radioprotection et de Surete Nucleaire (IRSN) for estimation of PWR fuel rod behavior during fast transients *e. g.* reactivity initiate accidents. The code is used for the analysis of integral tests carried out in NSRR and CABRI experimental facilities [4, 8].

FRAS fuel behavior analysis code is developed by Kurchatov Institute, Russia, for the analysis of fuel rod thermomechanical behavior in case of loss of coolant accidents and reactivity initiated accidents [9]. There are many other fuel behavior analysis computer codes such as TRANSURANUS which is developed by European Institute of Transuranium Elements (ITU) and FEMAXI developed by Japan Atomic Energy Agency (JAEA) and it is incorporated with improved mathematical representation of stress and strain [4, 10].

The examples of code modification and coupling includes the coupling of thermal hydraulic and fuel behavior analysis codes GENFLO-FRAPTRAN [11], modification of DYN3D code by including the simple fuel rod behavior models [12] and the development of DRACCAR and FRETA-B codes, *etc*. [4].

DESCRIPTION OF THEATRe CODE

Thermal hydraulic engineering analysis tool in real-time (THEATRe) is a computer simulation tool developed by GSE Power systems [4, 13]. The main design objectives of the code is the real time best estimate thermal hydraulic analysis of the nuclear power plant and it is a useful tool for developing the real time plant simulators for the operator training purposes. The THEATRe code is based on the numerical methodology and the physical models of RELAP5 MOD3 with some modifications which make the code suitable for real time applications. The state-of-the-art real time technology adopted in THEATRe code is its sophisticated matrix solver routine and the drift flux model [4, 13].

The THEATRe code can be applied for simulating the steady-state, transient and accident condition in nuclear power plant. The code is quite versatile for thermal hydraulic applications; however, the code approximate the fuel pin behavior, in that, as-fabricated fuel rod dimensions are utilized during the analysis. The code can be modified by incorporating the improved fuel pin behavior mathematical models for achieving more consistent results in steady state as well as in accident analysis.

Modifications in THEATRe code

In this study, THEATRe code is modified by adding important fuel behavior models which affects the heat transfer from fuel to the cladding. The modified version of the code is incorporated with the improved gas-gap heat conduction model, fuel rod thermomechanical behavior model, internal gas pressure model and the improved material properties correlations to simulate the burnup effects of the fuel. The gas-gap heat conduction model can evaluate the heat conduction through gas, radiation heat conduction and the solid-solid contact heat conduction due to fuel-cladding gap closure. The fuel rod thermomechanical model evaluates the updated fuel cladding gap by estimating the fuel and cladding deformation. The internal gas pressure model is used to calculate the gas pressure in the gap. The important material properties correlations are added in the modified version of the code including the burnup dependent fuel thermal conductivity. All the fuel behavior models and material properties correlations are obtained from FRAPTRAN code. The detail description of all the models is provided in the relevant sections of this paper.

Other than fuel rod behavior, the fuel to the coolant heat transfer is also affected by the heat transfer conditions at the fuel rod surface. The original THEATRe code is a comprehensive thermal hydraulic code and covers range of correlations for the heat transfer from heat structure surface to the coolant. The modified THEATRe code uses the same fuel rod surface heat transfer correlations/modes that are used in original THEATRe code. The heat transfer mode in original THEATRe code are the convection to noncondensable-water mixture, single-phase liquid convection at critical and supercritical pressure, single-phase liquid convection at subcritical pressure, subcooled nucleate boiling, saturated nucleate boiling, subcooled transition film boiling, saturated transition film , boiling subcooled film boiling, saturated film boiling, single-phase vapor convection, condensation when void is less than one, condensation when void equals one [4, 13].

As far as steady-state analysis is concerned, the modified THEATRe code is not designed for simulating the steady-state long-term burnup effects of the fuel rod; however, it can simulate the steady-state conditions of the reactor with fresh fuel. The modified code is designed for burnup dependent transient analysis, for that, the burnup dependent fuel rod initial conditions are required and these initial conditions can be obtained from the available data in literature or can be generated through steady-state fuel rod behavior analysis code such as FRAPCON. In this study, these initial conditions are obtained from the literature. The primary purpose of this study is the simulation of those fuel rod thermal and mechanical behaviors which are related to the fuel rod heat transfer during initial phase of RIA, but not the exact and detailed description of the complex fuel behavior phenomenon which are difficult to model in a thermal hydraulic code specially in a real time analysis code and separate special codes are required for that purpose such as FRAPTRAN.

The modified version of THEATRe is utilized to investigate the effect of fuel burnup on fuel rod thermal performance in steady-state and transient conditions. In order to investigate the transient fuel rod behavior, a hypothetical RIA is simulated by assuming a triangular power pulse of varying amplitude (relative to the nominal reactor power level) and pulse half width. Moreover, the expected fuel failure mechanisms are also investigated. The AP1000 reactor is considered for this analysis.

FUEL ROD BEHAVIOR MODELS

Thermal model in THEATRe code

The original THEATRe code evaluates the fuel and cladding temperatures by solving the heat conduction equation in radial direction. The assumption of one-dimensional (radial) heat transfer in original code is because of small fuel rod radial dimensions as compared with its axial length. The governing equation used in original THEATRe code for heat conduction in a cylindrical fuel rod is given by [4, 13]

$$\frac{\partial^2 T}{\partial r^2} \quad \frac{1}{r}\frac{\partial T}{\partial r} \quad \frac{q}{k} \qquad \frac{\rho C_{\mathrm{p}}}{k}\frac{\partial T}{\partial r} \tag{1}$$

where q''' is the thermal power generation density.

The original THEATRe code uses the quasi steady-state analytical method to calculate the temperature profile in heat structures. The same heat conduction equation of original THEATRe code mentioned in eq. 1 is utilized in modified THEATRe cod.

Gas-gap heat conduction

The fission heat generated in the fuel is conducted through fuel, gap and cladding to the coolant. The thermal properties of fuel and gas-gap are the main sources of thermal resistance for heat conduction from fuel to the cladding. The heat transfer coefficient (HTC) in the gap is mainly comprises of three terms including the gas heat conduction, the radiation heat conduction and the solid-solid contact heat condition due to gap closure. The gap heat transfer is given in eq. 2. Equation 3 represent the Ross-Stoute model for gas heat conduction terms and eq. 4 represent the Kreith correlation for radiation heat conduction term in eq. 2, respectively, [4, 10]

$$h_{\mathrm{gap}} \quad h_{\mathrm{g}} \quad h_{\mathrm{r}} \quad h_{\mathrm{s}} \tag{2}$$

$$h_{\mathrm{g}} \quad \frac{k_{\mathrm{gas}}}{d_{\mathrm{eff}} \quad d \quad g_{\mathrm{f}} \quad g_{\mathrm{c}}} \tag{3}$$

$$h_{\mathrm{r}} \quad \sigma F_{\mathrm{e}} F_{\mathrm{a}} (T_{\mathrm{fs}}^2 \quad T_{\mathrm{ci}}^2)(T_{\mathrm{fs}} \quad T_{\mathrm{ci}}) \tag{4}$$

The solid-solid contact heat conduction term in eq. 2 is calculated using correlations given in eq. 5 through eq. 7. The gas heat conduction term in eq. 3 is the main component of gap HTC when the fuel cladding gap is open [14]. This heat conduction through gas is a function of gas composition and the gap width.

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The radiation heat conduction term is significant in case of cladding ballooning [7]. The solid-sloid contact term is mainly a function of fuel-cladding contact pressure and the surface roughness [14]. The correlations used for computing gap HTC in modified THEATRe code are obtained from FRAPTRAN 1.5 code [4]

$$h_s \quad 0.4166\frac{k_m P_{rel} R_{mult}}{RE}, \text{ if } P_{rel} \quad 0.003 \tag{5}$$

$$h_s \quad 0.00125\frac{k_m}{RE}, \text{ if } 0.003 \quad P_{rel} \quad 9.0 \ 10^{\ 6} \tag{6}$$

$$h_s \quad 0.4166\frac{k_m P_{rel}^{0.5}}{RE}, \text{ if } P_{rel} \quad 9.0 \ 10^{\ 6} \tag{7}$$

where $E = \exp[5.738 - 0.528 \ln(\mathrm{Rf}\, a)]$, and a is a constant.

Fuel rod thermomechanical behavior

The evaluation of fuel rod thermomechanical behavior is important for estimation of changes in the fuel rod dimensions which occurs due to long-term fuel residence time in the reactor and also during transients/ accidents. The thermomechanical models calculate the fuel and cladding thermal expansion, stress induced cladding deformation and change in the gas-gap width. All the thermomechanical models used in modified code are the FRAPTRAN code models. Similarly, rigid fuel pellet assumption of FRAPTRAN code is employed in modified THEATRe code and stress induced fuel pellet deformation is not considered [4, 7]. The modified code provides more appropriate temperature distribution in the fuel rod according to the updated fuel, cladding and gap condition. The modified code updates the fuel rod dimensions and temperature distribution within the fuel rod at every time step.

The analysis of thermomechanical fuel rod behavior can be characterized according to the physical condition of the fuel rod gap width. If there is no physical contact between fuel pellet and cladding, this situation can be termed as open gap and the open gap solution method is employed. However, if the fuel pellet and cladding are in contact, the closed gap solution method is applied.

In open gap solution, the code evaluates the cladding deformation as a function of internal fuel rod pressure, external coolant pressure and the cladding temperature by assuming the cladding as a cylindrical shell. The physical state of the gas-gap (open or closed) is determined by comparing the radial displacement of the fuel and the radial displacement of the cladding due to external coolant pressure and internal fuel rod gas pressure in the gap. The modified THEATRe code utilizes eq. 8 to determine the open or closed state of the gap. If eq. 8 is not satisfied, the gap is open and the open gap solution is used. If eq. 8 is satisfied, the PCMI condition is reached and the closed gap solution is applied. The new gap width is calculated by using eq. 9

$$\delta R_f \quad \delta R_{cl} \quad \delta_r \tag{8}$$

$$\delta_{gap\ width} \quad \delta_r \quad \delta R_{cl} \quad \delta R_f \tag{9}$$

The set of equations for computing cladding deformation model are given in tab. 1. The total strain is computed using eq. 10 through eq. 12. Equation 10 through eq. 12 are the generalized form of Hooke's law and represent the stress-strain relations in incremental form. In these equations, the terms ε^p and $d\varepsilon^p$ represent the plastic strain at the end of previous load increment and the additional plastic strain increment during new load increment, respectively. The solution method adopted by Geelhood *et al*., 2014 is utilized to calculate the $d\varepsilon^p$. The main assumptions are the axisymmetric loading and deformation and uniform axial loading with no bending [4, 7] and eq. 13 and eq. 14 represent the equilibrium equations in this case. Equation 15 through eq. 17 represents the additional plastic strain increments based on effective stress and the Prandtl-Reuss flow rule [7].

The closed gap condition is termed as pellet cladding mechanical interaction (PCMI). In closed gap solution, the cladding deformation caused by internal rod pressure in open gap model is replaced with the radial displacement of cladding inner surface which is due to the pressure exerted by fuel pellet deformation on the cladding inner surface. The closed

Table 1. FRAPTRAN code clad deformation model employed in modified THEATRe code [4, 7]

Equation	No.
$\varepsilon_\theta \quad \frac{1}{E}(\sigma_\theta \quad \nu\sigma_z) \quad \varepsilon_\theta^p \quad d\varepsilon_\theta^p \quad \int_{T_0}^{T} \alpha_\theta dT$	(10)
$\varepsilon_z \quad \frac{1}{E}(\sigma_z \quad \nu\sigma_\theta) \quad \varepsilon_z^p \quad d\varepsilon_z^p \quad \int_{T_0}^{T} \alpha_z dT$	(11)
$\varepsilon_r \quad \frac{\nu}{E}(\sigma_z \quad \sigma_\theta) \quad \varepsilon_r^p \quad d\varepsilon_r^p \quad \int_{T_0}^{T} \alpha_r dT$	(12)
$\sigma_\theta \quad \frac{r_i P_i \quad r_o P_o}{t}$	(13)
$\sigma_z \quad \frac{r_i^2 P_i \quad r_o^2 P_o}{r_o^2 \quad r_i^2}$	(14)
$\sigma_e \quad \frac{1}{\sqrt{2}}[(\sigma_\theta \quad \sigma_z)^2 \quad \sigma_z^2 \quad \sigma_\theta^2]^{0.5}$	(15)
$d\varepsilon_i^p \quad \frac{3}{2}\frac{d\varepsilon_i^p}{\sigma_e} S_i$	(16)
$S_i \quad \sigma_i \quad \frac{1}{3}(\sigma_\theta \quad \sigma_z)$	(17)

gap solution is quite different and complex as compared with the open gap solution. The stresses in open gap solution are directly computed by internal and external pressures and the cladding temperature. However, in case of closed gap, the cladding stresses are the function of cladding inner surface displacement due to fuel thermal expansion. The cladding inner surface displacement can be determined by eq. 18. The cladding stresses (σ_θ and σ_z) can be computed in terms of cladding inner surface displacement, $u(r_i)$, by simultaneously solving eqs. 10-12, and 18. When the stresses are determined, the contact pressure can be computed using eq. 19

$$u(r_{\rm i}) = \bar{r}\varepsilon_\theta + \frac{t}{2}\varepsilon_{\rm r} \tag{18}$$

$$P_{\rm cont} = \frac{t\sigma_\theta + r_0 P_0}{r_{\rm i}} \tag{19}$$

Fuel rod internal gas pressure

In order to compute cladding deformation and heat transfer across fuel and cladding gap, the internal fuel rod gas pressure must be calculated appropriately. The rod internal pressure is mainly a function of the available free volume, the amount and type of gas(s) and its temperature. The fuel rod is axially divided into several small volumes and the temperature in each small volume can be assumed as constant [4, 7]. The fuel and cladding deformation model evaluate the change in the gap width in each small volume during steady-state and transient conditions. The plenum volume is assumed as constant during the transient. Temperature of the gas in the gap is computed by temperature model and the SCANAIR code assumption is employed, in that, the gas temperature (T_g) is assumed equal to the mean value of fuel surface temperature (T_{fs}) and the cladding inner surface temperature (T_{ci}) as given in [4, 8]

$$T_{\rm g} = \frac{T_{\rm fs} + T_{\rm ci}}{2} \tag{20}$$

During fast transient such as RIA, the gas(s) in the gap and in plenum does not reach to their thermal equilibrium. Therefore, the temperature of the upper and lower segments of the fuel rod can be assumed for the upper and lower plenum, respectively [8]. The change in gas pressure during transient is computed by eq. 21. Uniform pressure is assumed everywhere *i. e.* pressure instantaneously comes into equilibrium in the gas-gap and fuel rod plenums. The change in internal gas pressure is computed by evaluating the change in gap width and change in gas temperature. The change in the composition of gas in the gap is considered according to the burnup of the fuel and obtained from literature [4, 10]

$$P_{\rm g} = P_{\rm g,i}\frac{T_{\rm g}}{T_{\rm g,ini}}\frac{V_{\rm P} + V_{\rm g,ini}}{V_{\rm P} + \sum_{n=1}^{N} V_{\rm g,n}} \tag{21}$$

Material properties

The appropriate description and evaluation of material properties is significant for determining the heat conduction and temperature distribution in the fuel rod. The material properties correlations given in FRAPTRAN material properties hand book [4, 15] are utilized in modified THEATRe code. The important material properties employed in modified code are the thermal conductivities of fuel rod materials and gases in the gap, thermal expansion coefficients of fuel and cladding, fuel and cladding specific heat capacity and elastic modulus, *etc*.

Fuel burnup has a significant effect on its thermal conductivity. Fuel burnup causes degradation of fuel thermal conductivity due to irradiation damage to the fuel lattice and buildup of fission products. In order to consider the burnup effects on fuel thermal conductivity, the FRAPTRAN fuel thermal conductivity model for UO_2 fuel is selected which is adopted from modified Nuclear Fuels Industries (NFI) and Duriez/modified NFI models [4, 15]. The modified fuel thermal conductivity is given in eq. 22 and eq. 23

$$k_{95} = \frac{1}{A + BT + f(Bu) + [1 - 0.9\exp(-0.04Bu)]g(Bu)h(T)} + \frac{E}{T^2}\exp\left(-\frac{F}{T}\right) \tag{22}$$

$$h(T) = \frac{1}{1 + 396\exp\left(-\dfrac{6380}{T}\right)} \tag{23}$$

The burnup and fuel temperature are assumed as radially averaged values.

AP1000 MODEL AND RIA SIMULATION

The AP1000 reactor is modeled in THEATRe code as per the nodalization diagram given in fig. 2. The AP1000 fuel assembly is composed of cylindrical UO_2 fuel pins. The UO_2 fuel pellet is covered with ZIRLO cladding and the fuel-cladding gap is filled with helium gas.

The steady-state results of AP1000 calculated from THEATRe model are compared with the data reported by Westinghouse in design control documents (DCD) [16]. The THEATRe code results are closely agreed with the reference results as shown in tab. 2. The main design parameters of AP1000 fuel rod are considered the same as given in AP1000 design control documents [16] and mentioned in tab. 3.

In order to investigate the effect of RIA on fuel rod behavior at various fuel burnups in a pressurized water reactor under power operation, hypothetical test cases of power pulse insertion are considered in AP1000 reactor at nominal power, pressure and coolant mass flow rate.

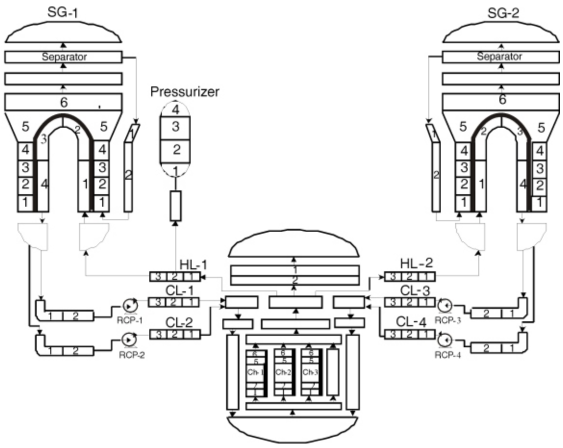

Figure 2. AP1000 nodalization in THEATRe code

Table 2. Comparison of AP1000 parameters

Parameters	THEATRe
Core thermal power [MW]	3410
Coolant volume flow per loop [m^3s^{-1}]	9.98
RCS pressure [MPa]	15.52
Core inlet temperature [K]	554.22
Core outlet temperature [K]	594.75
Core average temperature [K]	574.4
SG secondary pressure [MPa]	5.602
SG feed water temperature [K]	499.82

Table 3. Design parameters of AP1000 fuel rod

Parameters	DCD value [16]
Average linear power [kWm^{-1}]	19.22
Fill gas initial pressure [MPa]	2
Active fuel rod length [cm]	426.72
Fuel pellets diameter [cm]	0.819
Diameter gap [cm]	0.0165
Clad thickness [cm]	0.0572
Pitch [cm]	1.260
Fuel rod outside diameter [cm]	0.950

A hypothetical RIA is simulated by considering a triangular power pulse of variable pulse height (relative to the full power reactor operating conditions) and pulse width. The burnup levels considered in this analysis are 0 GWD/MTU*, 6.4 GWD/MTU and 21.3 GWD/MTU, which corresponds to fresh fuel, low and medium burnup fuels respectively. The burnup dependent steady state and transient fuel rod behavior analysis requires the burnup dependent initial conditions as input in the code. The reference gap width, gas composition and cavity pressure are obtained from the fuel rod behavior analysis of AP1000 reactor performed by Yu *et al*., 2012 [10].

RESULT AND DISCUSSIONS

Steady-state comparison

The comparison of steady-state fuel rod parameters at different burnups is given in fig. 3 through

*GWD/MTU means gigawatt-days per metric tonne of uranium

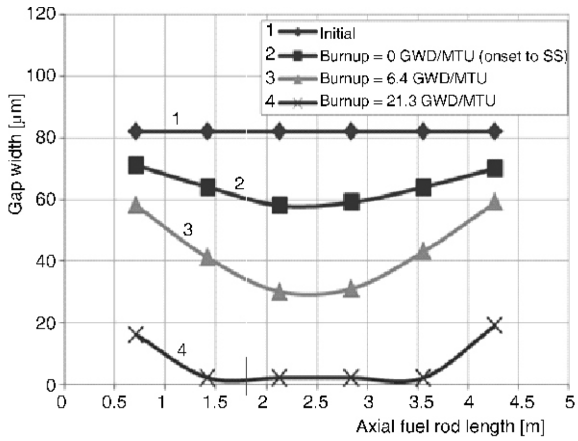

Figure 3. Steady-state gap width at fuel rod axial locations

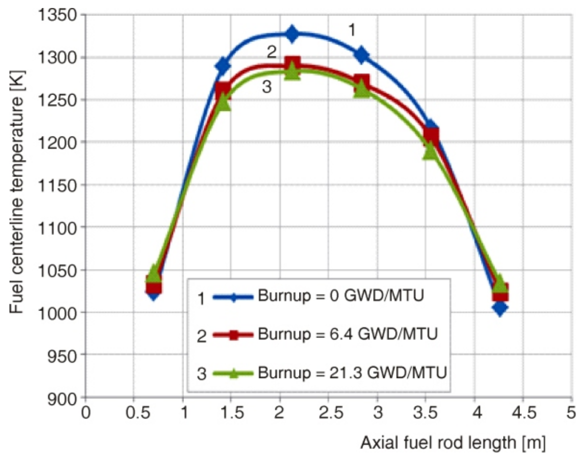

Figure 4. Steady-state fuel centerline temperature at fuel rod axial locations

fig. 6. The gap width along fuel rod axial length at different burnups is given in fig. 3. In fig. 3, the straight line *i. e.* the same gap width at each axial position (blue line) represents the fresh fuel cold state gap width. However, the other curves represent the gap widths of fresh and burned fuel rods at average normal reactor operating condition. This gap width data is obtained from literature [10] and used as input in this work.

At 21.3 GWD/MTU, the gap width at axially central locations of fuel rod is near to closed gap width. The axial distribution of fuel centerline temperature is given in fig. 4. This axial fuel temperature distribution is caused by the near-cosine shape axial power distribution in the fuel. In fig. 4, it is shown that the fuel centerline temperature at upper and lower portion of the fuel rod is slightly higher in case of relatively higher burnup fuel. However, at central portion of the fuel, this temperature is lower for relatively higher burnup fuel due to a decreased gap width which affects the gas-gap conductance. The gas-gap conductance is affected by gap width and gas composition in the gap. The fuel burnup causes decrease in gap width and helium mole fraction. These parameters have opposite effects on gas-gap conductance *i. e.* reduction in gap width increases the gap conductance and reduction in helium mole fraction decreases the gap con-

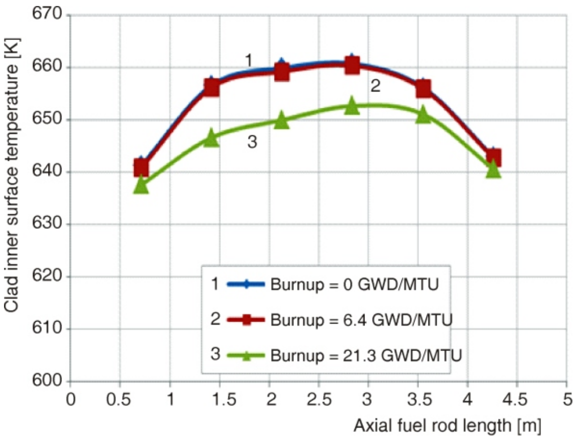

Figure 5. Steady-state cladding inner surface temperature at fuel rod axial locations

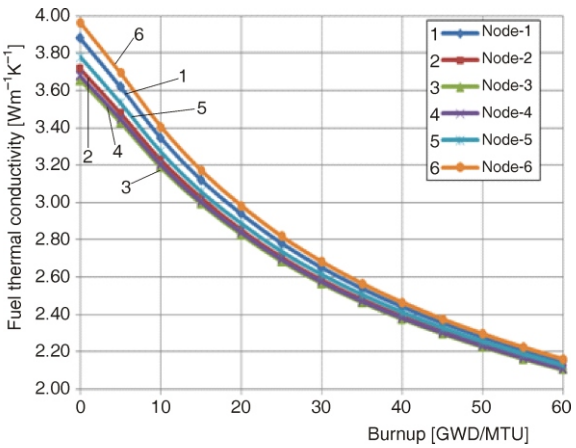

Figure 6. Steady-state fuel thermal conductivity *vs.* fuel burnup [17]; node-1 and node-6 represent upper and lower segments of fuel rod, respectively

ductance. When gap closure occurs at high burnups, the gap closure effect dominates and the heat transfer coefficient (HTC) in gas-gap increases significantly. The axial distribution of cladding inner surface temperature is shown in fig. 5. The high burnup fuel rod has lower cladding inner surface temperature. The axial fuel rod length is divided into six nodes. Fuel thermal conductivity is inversely proportional to fuel burnup. Figure 6 shows the fuel thermal conductivity degradation as a function of burnup along axial fuel rod length at each axial node. This thermal conductivity is calculated at steady state operation at corresponding fuel burnup. With increase in burnup, the fuel thermal conductivity decreases and the thermal conductivity difference between the fuel rod axial locations also decreases, which indicates the dominance of fuel burnup effect on fuel thermal conductivity over the fuel temperature.

Transient fuel rod behavior under RIA

A hypothetical RIA is simulated by considering a triangular power pulse with varying amplitude and pulse width as describe in *AP1000 MODEL*. The effect of fuel burnup, power pulse height and the pulse width on fuel rod parameters has been investigated in this study.

Effect of pulse height

In this work, the focus of analysis is to see the fuel behavior. Therefore, hypothetical RIA conditions considered for analyzing the fuel failure mechanism at different burnups. The effect of pulse height is investigated by inserting a triangular power peak of variable amplitude relative to the nominal steady-state reactor power but with fixed half width of 0.4 s. In order to analyze fuel failure mechanisms, reactor shutdown at overpower is not considered. The immediate effect of power pulse insertion is the rapid rise of fuel temperature; therefore, the fuel centerline temperature of axially central portion of the fuel rod is investigated. The results at 0 GWD/MTU and 6.4 GWD/MTU fuel burnups are shown in fig. 7. The rise in fuel centerline temperature is obvious with increase in reactor power due to power pulse insertion. For power pulse height (PH) of 6 and 10 relative to nominal power, the increase in fuel centerline temperature is nearly same for both the cases of zero GWD/MTU and 6.4 GWD/MTU. At zero GWD/MTU with power pulse height of 20, the peak fuel centerline temperature becomes greater than 2800 K, which is near to fuel melting temperature and no gap closure at any fuel rod axial location has been observed. The other observed fuel rod parameters (not mentioned in fig. 7) includes as follows.

The gap closure starts as the power pulse height increase beyond 20 relative to the nominal reactor power. In case of 6.4 GWD/MTU, the gap closure starts when the pulse height increases beyond 10 and for fuel with burnup of 21.3 GWD/MTU, the gap closure starts as the pulse height increase beyond 6 relative to the nominal reactor power.

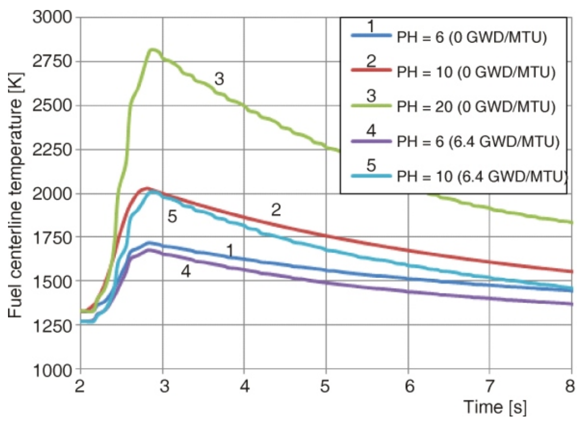

Figure 7. Fuel centerline temperature with variable power pulse height at pulse half width of 0.4 s

Effect of pulse width

In order to investigate the effect of power pulse width, analysis is performed by varying power pulse half width by 0.4 s, 0.5 s, and 0.75 s while keeping the power pulse height as 6 relative to the nominal reactor power. The analysis is performed by considering the 6.4 GWD/MTU fuel burnup and results are mentioned in fig. 8 and fig. 9. The results show that by increasing power pulse width, the peak fuel rod temperature also increases. No gap closure at any axial fuel rod location is observed. The behavior of cladding inner surface temperature is mentioned in fig. 9. The effect of pulse width with fuel burnup of 21.3 GWD/MTU is also investigated and it is observed that the gap closure starts when the pulse half width increase beyond 0.4 s.

Effect of fuel burnup

The effect of fuel burnup on fuel rod temperatures is investigated by considering the power pulse height of 6 relative to the nominal reactor power and the pulse half width of 0.4 s. The average linear power is same in all cases. In fig. 10, the fuel centerline temperatures at axially central and upper segments of the fuel rod are compared. It is to be noted that in upper portion of the fuel rod, the initial (steady-state) fuel centerline temperature is relatively higher at relatively higher burnup *i. e.* 21.3 GWD/MTU. However, at central segment of fuel rod, the steady-state fuel centerline temperature is lower in higher burnup case due to the heat conduction through gap which dominates the effect of fuel thermal conductivity degradation due to burnup. Similarly, the increase in temperature after power peak insertion has similar trend.

Cladding hoop stress is a function of rod internal pressure, cladding thickness and coolant pressure. The calculated hoop stress is nearly equal at each axial node due to the identical internal rod pressure and external coolant pressure. Comparison of hoop stresses at central node of the fuel rod for all three cases is given in fig. 11. The steady-state cladding hoop stress is lower in case of the fuel rod with comparatively high burnup due to relatively high internal rod pressure. During course of the transient, variation in coolant pressure, cladding deformation and the increase in the rod internal pressure determine the shape of the curve mentioned in fig. 11. The increase in hoop stress is also less in high burnup fuel due to high initial gas pressure and increase in the gas pressure during the transient.

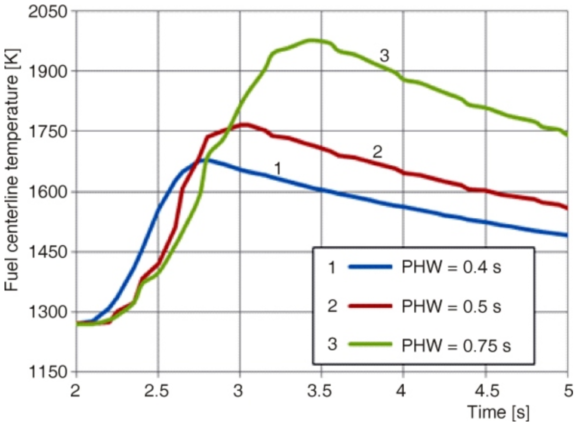

Figure 8. Fuel centerline temperature with variable pulse half width at 6.4 GWD/MTU and PH = 6

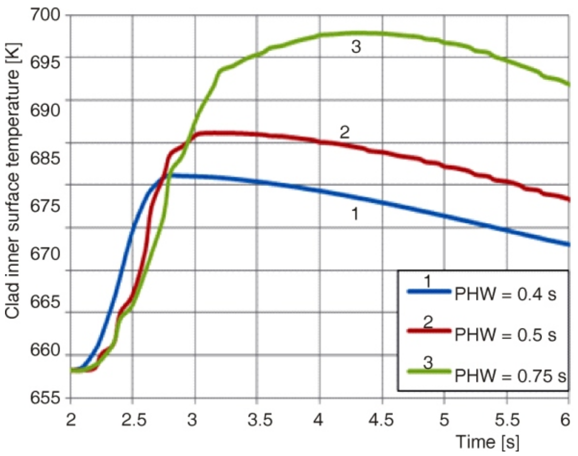

Figure 9. Variation of cladding inner surface temperature with pulse half width at 6.4 GWD/MTU and PH = 6

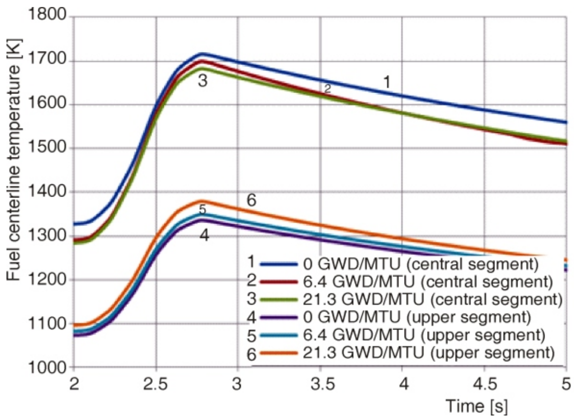

Figure 10. Fuel centerline temperature *vs.* fuel burnup during RIA

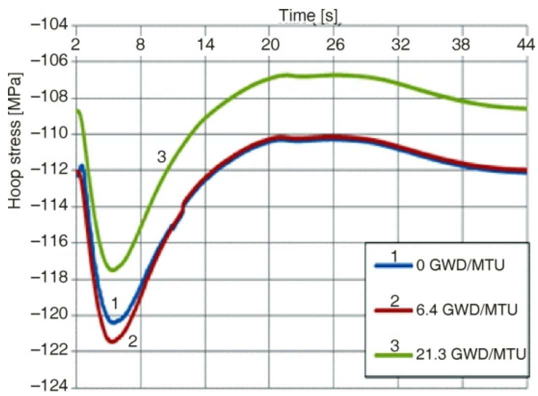

Figure 11. Comparison of hoop stresses at central node of fuel rod for all three cases

Table 4. Fuel rod behavior during assumed RIA as calculated by modified TEATRe code

Parameters	$T_{c,max}$ [K]	$T_{f,max}$ [K]	$T_{clad,max}$ [K]	Specific remarks on fuel behavior
Burnup = 0				
Power pulse height = 6	1715.35	809.21	679.53	No gap closure occurs
Power pulse height = 10	2027.09	820.02	696.18	No gap closure occurs
Power pulse height = 20	>2800	1135.85	1114.37	Fuel melting expected $H > 20$, beginning of gap closure
Burnup = 6.4 GWD/MTU				
Power pulse height = 6	1678.05	760.84	681.16	No gap closure occurs
Power pulse height = 10	2011.49	781.37	696.69	No gap closure occurs If $H > 10$, beginning of gap closure
Power pulse height = 20	>2500	1591.86	1591.86	
Burnup = 21.3 GWD/MTU				
Power pulse height = 6	1682.21	679.14	672.46	If $H > 6$, beginning of gap closure

$T_{c,max}$ – maximum fuel centerline temperature, $T_{f,max}$ – maximum fuel surface temperature, $T_{clad,max}$ – maximum cladding surface temperature. Pulse height (relative to the full power reactor operating conditions)

Assessment of fuel ROD failure during RIA

The fuel rod behavior results observed during all RIA cases considered in this work have been compiled and given in tab. 4. Fuel rod temperatures and fuel behavior mentioned against each case are at axially central portion of the fuel rod. Following can be summarized from the results mentioned in tab. 4.

– With increase in power pulse height (pulse half width = 0.4 s), the fuel failure in fresh fuel is expected due to increase in fuel temperature, which may reach to fuel melting point before start of gap closure phenomena. It can be seen in tab. 4, with fresh fuel at pulse height of 20.
– It has been observed that, at any burnup, the gap closure phenomena initially occur at high power axial locations in the fuel rod. At gap closure, the fuel surface and cladding inner surface temperature becomes nearly identical as given in tab. 4 at 6.4 GWD/MT fuel burnup and PH = 20.
– The fuel failure at 21.3 GWD/MTU fuel burnup is expected by cladding failure. In higher burnup fuels, the gap closure occurs much earlier than the fresh and low burnup fuels. As mentioned in tab. 4, the gap closure at 21.GWD/MTU fuel burnup starts when the pulse height increases beyond 6. At gap closure, the heat transfer becomes better, however, the contact pressure may exceed the cladding yield strength and cladding failure due to PCMI is expected.

CONCLUSIONS

The consistent simulation of accidents and transients is important for reactor safety analysis. In this work, fuel rod thermal performance in steady-state and transient conditions has been analyzed at different fuel burnup conditions by modifying THEATRe code with inclusion of fuel behavior models. AP1000 reactor is considered for this analysis. For transient analysis, a hypothetical RIA is simulated. A simple triangular power pulse of variable height relative to nominal reactor power level and with variable half width at different burnups is assumed.

This work aimed at the simulation of those fuel rod thermal and mechanical behaviors which affect the heat transfer during initial phase of RIA, but not at the exact description of complex phenomenon of fuel rod behavior which is difficult to model in thermal hydraulic code and separate dedicated fuel rod behavior codes are available for that purpose.

The comparison of steady-state fuel centerline temperature at different axial locations of the fuel rod shows that at upper portion, the initial (steady-state) fuel centerline temperature is relatively higher at higher burnup *i. e.* 21.3 GWD/MTU. However, at central segment of the fuel rod, the steady-state fuel centerline temperature is lower in higher burnup case due to heat conduction through gap which dominates the effect of fuel thermal conductivity degradation due to fuel burnup.

The results of transient RIA analysis indicate that the fuel failure mechanisms of fresh and higher burnup fuels are different. The fuel failure in fresh fuel is expected due to fuel melting as fuel temperature increases with increase in pulse energy (pulse height). However, at relatively higher burnups, the fuel failure is expected due to cladding failure due to strong pellet clad mechanical interaction (PCMI) which may cause cladding failure when the contact pressure increases beyond cladding yield strength.

The new fuel rod behavior models in THEATRe code are capable of estimating the gas-gap behavior during transient conditions which is relevant to the consistent calculations of fuel rod parameters during transient conditions in a thermal hydraulic code. However, these models need further improvement in order to account for complex phenomena of fuel behavior and for the analysis of high burnup fuels.

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AUTHOR CONTRIBUTIONS

Theoretical analysis was carried out by A. Nawaz and A. Hussain, code modification and simulation work was carried out by A. Nawaz. The analysis on results was carried out by all the authors. The manuscript was written by A. Nawaz in co-ordination with Y. Hidekazu and M. Yang.

NOMENCLATURE

C_p – heat capacity of the coolant, [$\mathrm{Jkg^{-1}K^{-1}}$]
d – open fuel-cladding gap size, [m]
d_{eff} – effective gap width, [m]
$d\varepsilon_i^p$ – additional plastic strain increments at new load
E – modulus of elasticity, [$\mathrm{Nm^2}$]
F_a – configuration factor = 1.0
F_e – emissivity factor
g_c – cladding inside temperature jump distance, [m]
g_f – temperature jump distance at fuel surface, [m]
HTC – heat transfer coefficient
h_{gap} – total gap conductance, [$\mathrm{Wm^{-2}K^{-1}}$]
h_{gas} – conductance through gas in gas-gap, [$\mathrm{Wm^{-2}K^{-1}}$]
h_r – conductance by radiation, [$\mathrm{Wm^{-2}K^{-1}}$]
h_s – conductance by fuel clad solid-solid contact, [$\mathrm{Wm^{-2}K^{-1}}$]
K_m – mean thermal conductivity, [$\mathrm{Wm^{-1}K^{-1}}$]
k – thermal conductivity, [$\mathrm{Wm^{-2}K^{-1}}$]
k_{gas} – gas thermal conductivity, [$\mathrm{Wm^{-2}K^{-1}}$]
PH – pulse height
PHW – pulse half width
P_i – internal fuel rod pressure, [$\mathrm{Nm^{-2}}$]
P_o – coolant pressure, [$\mathrm{Nm^{-2}}$]
P_{rel} – ratio of interfacial pressure to cladding Meyer hardness
R – effective roughness, [m]
r_i – cladding inside radius, [m]
r_o – cladding outside radius, [m]
T – current cladding temperature, [K]
T_0 – reference temperature, [K]
T_{fs} – fuel pellet outside temperature, [K]
T_{ci} – cladding inside temperature, [K]
t – cladding thickness, [m]

Greek symbols

α – thermal expansion coefficient, [$\mathrm{K^{-1}}$]
ε_z – axial strain
ε_r – radial strain
ε_i^p – plastic strain at the end of last load increment
ε_θ – hoop strain
θ_z – axial stress, [$\mathrm{Nm^{-2}}$]
ν – Poisson's ratio
ρ – density, [$\mathrm{kgm^{-3}}$]
σ – Stefan-Boltzmann constant
σ_e – effective stress, [$\mathrm{Nm^{-2}}$]
σ_θ – hoop stress, [$\mathrm{Nm^{-2}}$]

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АНАЛИЗА ТЕРМИЧКОГ ПОНАШАЊА ГОРИВА PWR РЕАКТОРА ТОКОМ АКЦИДЕНТА ИЗАЗВАНОГ РЕАКТИВНОШЋУ ПРИ РАЗЛИЧИТИМ САГОРЕВАЊИМА, ПРИМЕНОМ THEATRe КОДА

Озрачивање и сагоревање горива ствара геометријске и димензионалне промене у горивном елементу што има утицај на његову термичку отпорност и коначно делује на понашање горивног елемента током стационарног стања и прелазних режима. Доследна анализа термичких перформанси горивног елемента од суштинског је значаја за прецизну процену сигурности реактора у прелазним режимима и током акцидената. У овом раду, спроведена је анализа термичких перформанси PWR горивног елемента у стационарном стању и прелазним режимима при различитим сагоревањима. Анализа је изведена применом термохидрауличног програмског пакета THEATRe. Програмски пакет допуњен је додавањем модела понашања сагоревања горивног елемента. Оригинални програмски пакет користи фабричке димензије горивног елемента током стационарног стања и прелазних режима, које се могу изменити ради спровођења потпуније анализе сигурности реактора. AP1000 реактор сматра се референтним реактором за овакву анализу. Испитан је ефекат сагоревања на параметре горивног елемента у стационарном стању. За анализу прелазног режима, хипотетички акцидент изазван реактивношћу симулиран је троугластим импулсом снаге промењљиве висине импулса (у односу на радне услове реактора при пуној снази) и ширине импулса, при различитом сагоревању горива које одговара свежем гориву, мало и средње утрошеном гориву. Испитана је зависност температуре горивног елемента од утицаја снаге, висине импулса, ширине импулса и сагоревања горива. Резултати анализе акцидента изазваног реактивношћу показују да су механизми отказивања горивног елемента различити за свеже гориво и горива различитог нивоа истрошености. Отказ горивног елемента свежег горива очекиван је услед топљења горива, јер температура горива расте са порастом импулса енергије (висине импулса). Међутим, при релативно високим нивоима сагоревања, отказ горивног елемента очекиван је услед пропадања кошуљице изазване јаким механичким интеракцијама горивних пастила и кошуљице, јер је притисак контакта већи од чврстине кошуљице.

Кључне речи: акцидент изазван реактивношћу, термичко понашање горивног елемента, деформација кошуљице, AP1000 реактор, THEATRe програмски пакет