Simulation of fuel rod behavior under reactivity-initiated accident conditions

Moustafa S. El-Koliel, Attya A. Abou-Zaid and A. A. El-Kafas
Reactors Dept., Nuclear Research Center, Atomic Energy Authority, P. O. Box 13759 - Cairo, Egypt

The Reactivity-Initiated Accident (RIA) in Pressurized Water Reactor (PWR) system has been modeled and simulated. The model was developed to describe the transient thermal behavior of burned PWR fuel during RIA and to predict the possibility of the rod failure in the mode of cladding melting. The modeling of RIA is divided into three subsystems; reactor kinetics, reactor core and cooling system thermal-hydraulics, and the release of Fission Products (FPs) from UO_2 fuel rods of PWR. The detailed temperature history determined from the thermal-hydraulic calculations, and the inventories of FPs calculated by ORIGEN2 code, have been used for the determinations of FPs release during the proposed transient. This paper provides a real-time lumped model for the release of fission products from UO_2 under RIA conditions. All the models mentioned above have been assembled in one unified computer program, where the reactor response under RIA was represented by the generated power, the surface and centerline fuel temperature, fuel enthalpy and release rate as well as fractional release of some FPs at two different average burnup values (10 and 45 MWd/kgU).

تم عمل نموذج رياضي لدراسة السلوك الحراري المفاجيء لوقود مفاعلات الماء المضغوط في ظروف حادثة تنشأ عن زيادة الفاعلية (RIA) بالإضافة إلى التنبؤ بامكانية حدوث عطب بقضبان الوقود أو انصهار الغلاف الواقي للوقود. النموذج المصمم لهذا الحادث (RIA) ينقسم إلى ثلاثة أقسام فرعية هي كينتيكا المفاعلات، قلب و أنظمة التبريد الهيدروليكية والحرارية للمفاعل وكذلك تسرب نواتج الانشطار من قضبان الوقود المستخدم في مفاعلات الماء المضغوط. لقد تم حساب درجات ومخزون نواتج الانشطار واستخدامها في حساب النواتج الانشطارية المنبعثة من الوقود حال نشوء الحادث. تم تجميع كل النماذج السابق ذكرها في برنامج محاكاه حيث يتم تمثيل وعرض أداء المفاعل عند هذا الحادث من خلال معاملات قدرة المفاعل، درجات حرارة قلب وسطح قضبان الوقود النووى، إنثالبي الوقود، معدلات انبعاث نواتج الانشطار عند نسبتى احتراق ١٠، ٤٥ ميجاوات يوم/كجم يورانيوم.

Keywords: Reactivity-initiated accident (RIA), Fuel performance, Fuel enthalpy, Core inventory, Reactor kinetics

1. Introduction

The operation of nuclear power plant requires an understanding of the plant response to both expected and unlikely transients. Recognizing the fact that the large inventories of radioactive FPs in the irradiated fuel are the principal hazards associated with nuclear reactors accidents. Considerable efforts have been devoted to determine the source term, and the potential of release of these materials to the environment under a variety of accident conditions [1,2]. Many out-of-pile post irradiation experiments [3-5] indicate that for temperatures between 1000 and 2180 °K, fission products behavior is dominated by diffusional release of volatile Fission Products (FPs), xenon, krypton, cesium, iodine and tellurium from ruptured fuel rods. Very little release of the medium- and low-volatile FPs is expected at these temperatures. The FPs released from the fuel matrix through inta- and intergranular diffusion, surface vaporization, and cladding rupture are macroscopically transported by the steam-hydrogen flow through the reactor core and primary system.

An important postulated accident in the design and licensing of a nuclear reactor is the reactivity-initiated accident Reactivity-Initiated Accident (RIA). Reactor events that can initiate the RIA include control rod ejection/rod drop, steam line rupture and cold water addition. In general, fuel licensing analyses design bases using a bounding control rod ejection/rod drop accident. Such an accident includes a prompt critical power excursion with reactor power levels reaching 10 to 100 times rated

power in a less than one second before the neutronic parameters, Doppler and moderator coefficients terminate the events [6].

A control rod ejection accident has been analyzed in this work. The rod ejection accident is defined as an assumed failure of a control rod mechanism pressure housing such that the reactor coolant system pressure would eject the control rod and drive shaft to the fully withdrawn position. The consequence of this mechanical failure is a rapid reactivity insertion. The resulting increase of the reactor power is determined by the design features of the core; namely the temperature coefficients of the reactivity of the fuel (Doppler effect) and the moderator. The transient depends on many details such as temperatures in the reactor core (e.g. cold or hot conditions), the control rod and fuel-loading pattern and the insertion of the other control rods.

From the above discussion, the failure behavior of Pressurized Water Reactor (PWR) fuel rods is one of the important subjects to be studied for LWR safety under the RIA conditions. An exact prediction of the fuel material temperature requires simultaneous determination of the neutronic and temperature fields. An accurate description of the temperature distribution in the fuel rods, and the FPs inventory at the time of the accident is needed for the release fraction and fractional release rate calculations. So, the main purpose of the model is to analyze the transient thermal behavior of a PWR fuel rod and predict the possibility of its failure. The inventory of the reactor core has been calculated using ORIGEN2 code [7], for two different average fuel burnup values. The calculations have been performed for PWR UO_2 fuel of 3.2% enrichment.

A computer program was written to solve all the simultaneously equations set describing the model. Many of the mechanical and thermo-physical properties of the fuel are taken from MATPRO-11 [8]. To validate the results, finite-difference and Runge-Kutta methods are used to solve the point kinetic equation.

2. Outline of the proposed mathematical model

2.1. Reactor kinetics and delayed neutrons model

The kinetic equations simulate power generation within the nuclear fuel. Standard time-dependent point kinetics, based on neutron conservation, are assumed [9], A coupled set of first order differential equations defining the total power *P(t)* and the power due to delayed-neutron precursors concentration *C(t)* as functions of time will be as follows:

$$\frac{dP(t)}{dt} = \frac{\rho(t)-\beta}{\Lambda} P(t) + \sum_{i=1}^{6} \lambda_i C_i(t), \quad (1)$$

$$\frac{dC_i(t)}{dt} = \frac{\beta_i}{\Lambda} P(t) - \lambda_i C_i(t), \quad (2)$$

where:

t = time (s),

β_i = effective delayed neutron fraction for the i-th group,

β = effective delayed neutron fraction,

Λ = prompt neutron generation time (s),

λ_i = *i*'th delayed neutron group decay constant (s^{-1}),

$\rho(t)$ = net reactivity $= \frac{k-1}{k} = \frac{\delta k}{k}$, and

k = effective neutron multiplication factor.

During RIA the reactivity ρ varies due to externally inserted reactivity, ρ_{ext}, and feedbacks mainly due to temperature variations. This may be expressed as:

$$\rho(t) = \rho_{ext}(t) + \rho_f(t) + \rho_m(t), \quad (3)$$

$\rho_f(t)$, $\rho_m(t)$ = Doppler and moderator feedback reactivity, respectively.

Assuming the reactor has been operated at the initial power level P_o before the transient occurrence, the values of C_i at t = 0 are obtained in the usual way by setting its derivative equal to zero at the initial power level P_o as follows:

$$C_i(0) = \frac{\beta_i}{\lambda_i \Lambda} P_o \quad i = 1,2,\ldots,6. \tag{4}$$

Since the transient is assumed to occur when the reactor is operating in the steady-state, it becomes better to represent the reactor kinetic equations in their normalized forms [10]:

$$P^* = \frac{P - P_o}{P_o}$$
$$C^*_i = \frac{C_i - C_{io}}{C_{io}} \; i = 1,2,\ldots,6\,, \tag{5}$$

where:
P_o is the steady-state power (MW),
P is the transient power (MW),
P^* is the normalized power (MW),
C_{io} is the steady-state power due to neutron precursor of group i (MW),
C_i is the transient state power due to neutron precursor of group i (MW), and
C^*_i is the normalized power due to neutron precursor of group i (MW).

By solving for P and C and substituting in eqs. (1) and (2) the following equations are obtained.

$$P_o \frac{dP^*(t)}{dt} = \frac{\rho(t) - \beta}{\Lambda}\left(P_o P^* + P_o\right) + \sum_{i=1}^{6} \lambda_i \left(C_{io} C^*_i + C_{io}\right), \tag{6}$$

$$C_{io} \frac{dC^*(t)}{dt} = \frac{\beta_i}{\Lambda}\left(P_o P^* + P_o\right) - \lambda_i \left(C_{io} C^*_i + C_{io}\right), \tag{7}$$

by using eq. (4), the last equations are reduced to their normalized forms as follow:

$$\frac{dP^*(t)}{dt} = -\frac{\beta_i}{\Lambda} P^*(t) + \frac{1}{\Lambda}\sum_{i=1}^{6} \beta_i C^*_i(t) + \frac{1}{\Lambda}\rho(t) + \frac{1}{\Lambda}\rho(t) P^*(t), \tag{8}$$

$$\frac{dC^*_i(t)}{dt} = \lambda_i P^*(t) - \lambda_i C^*_i(t) \; i = 1,2,\ldots,6, \tag{9}$$

the initial conditions of eqs. (8) and (9) are:

$$P^*(t=0) = 0, \; C^*_i(t=0) = 0. \tag{10}$$

The total power $P_t(t)$ is equal the kinetics power $P(t)$ in addition to the FPs decay power. Constant decay power (6.73% of the initial power level) is assumed as long as a plant scram does not occur. Following a scram, this decay heat fraction is given by [11].

$$P_d(t) = \{0.1[(\tau - \tau_S + 10)^{-0.2} - (\tau + 10)^{-0.2} + 0.87(\tau + 2\times 10^7)^{-0.2} - 87(\tau - \tau_S + 2\times 10^7)^{-0.2}]\} \times P_o\,. \tag{11}$$

2.2. Thermal -hydraulic balance equations

A lumped parameter model is an adequate representation of the reactor core, where a single temperature node in the fuel and one in the surrounding coolant is used. Performing a heat balance on the reactor fuel and coolant yields the following state eqs. [12].

$$M_f C_{pf} \frac{dT_f(t)}{dt} = \eta P - U f_c \left(T_f - T_m\right), \tag{12}$$

$$M_c C_{pc} \frac{dT_c(t)}{dt} = Q_{fc} + (1-\eta)P - W_{core} C_{pc} (T_{out} - T_{in})\,, \tag{13}$$

where:

M_f, M_c	is the total mass of fuel and coolant respectively (Kg),
C_{pf}, C_{pc}	is the specific heat of fuel and coolant respectively (MJ/Kg.K),
$T_f(t)$, $T_c(t)$	is the average temperature of fuel and coolant respectively (K),
$T_{out}(t)$, $T_{in}(t)$	is the reactor coolant outlet and inlet temperatures (K); 2 T_c = T_{out} + T_{in}
W_{core}	is the core mass flow rate (Kg/s),
Q_{fc}	is the heat flow to the coolant from fuel (MW) = U_{fc} (T_f - T_c), and
U_{fc}	is the overall heat transfer coefficient between fuel and

coolant (MW/°K), by using the Dittus-Boalter correlation [13].

$$Nu = 0.023\, Re^{0.8}\, Pr^{0.4}. \quad (14)$$

The heat transfer coefficient is obtained as follows,

$$U_{fc} = 0.023\, \frac{K_c}{D_e}\, Re^{08}\, Pr^{0.4}, \quad (15)$$

where:
Nu is the Nasselt number = $U_{fc}\, D_e / K_c$,
Re is the Reynolds number = $W_{core}\, D_e / \mu A$,
Pr is the Prandtl number = $C_p\, \mu / k_c$,
D_e is the equivalent diameter of the flow channel (m),
μ is the dynamic viscosity of the coolant (kg/m.s)
ρ_c is the density of the coolant (kg/m^3), and
A is the cross sectional area of the channel (m^2).

The radial, $T_f(r)$ and surface fuel, T_{fs} temperatures are given as follows [12]:

$$T_f(r) = T_f + \frac{q'}{4\pi k_f}\left[1 - \left(\frac{r}{R}\right)^2\right] for \quad 0 \le r < R, \quad (16)$$

$$T_{fs} = T_f + q'/8\pi k. \quad (17)$$

The heat in terms of the linear heat rate can be represented as:

$$q' = \pi R^2 q'''. \quad (18)$$

Also, the linear heat rate can be represented in terms of the gap heat transfer coefficient h_g and the temperature drop across the fuel-cladding interface is,

$$q' = 2\pi R h_g \left[T_f - T_{ci}\right]. \quad (19)$$

The radial and outer surface of the clad temperatures are calculated as follows:

$$T_c(r) - T_{ci} = \frac{q'}{2\pi k_{cl}}\, \ell n\left(\frac{r}{a}\right) \; for \quad a < r \le b, \quad (20)$$

$$T_{ci} - T_{co} = \frac{q'}{2\pi k_{cl}}\, \ell n\left(\frac{b}{a}\right). \quad (21)$$

2.3. UO_2 enthalpy model

The enthalpy data of UO_2 [14] were fit to the equations:

For 293 °$K \le T \le$ 2670 K

$$H^o(T) - H^O(298.15) = C_1 \vartheta \left[\left(e^{\vartheta/T} - 1\right)^{-1} - \left(e^{\vartheta/T^*} - 1\right)^{-1}\right] + C_2\left[T^2 - \left(T^*\right)^2\right] + C_3 k_1\left(Te^{-E_a/k_1T} - T^* e^{-E_a/k_1T^*}\right), \quad (22)$$

where:
T^* = 298.15 °K
C_1 = 78.215 J mol^{-1} K^{-1},
C_2 = 3.8609×10^{-3} J mol^{-1} K^{-2},
C_2 = 3.4250×10^{8} J mol^{-1} eV^{-1},
E_a = 1.9105 eV,
ϑ = 516.12 K.
k_1 = 8.6144×10^{-5} eV K^{-1} (Boltzmann's constant);

For 2670 °K $\le T \le$ 3120 K;

$$H^o(T) - H^o(298.15) = 0.16704T - 281.342. \quad (23)$$

The failure of the fuel rod is controlled by several factors. The three significant independent variables are fuel burnup, oxide thickness and pulse width. Based on the published data, the failure enthalpy correlation is derived as follows [15]:

$$H_f = 156.6 - 0.774 \times OT - 1.076 \times BU + 29.41 \times log(PW), \quad (24)$$

where H_f is the failure enthalpy (cal/g), OT is the oxide thickness (μm), BU is the fuel burnup (GWd/tU) and PW is the pulse width in terms of Full-Width at Half-Maximum (FWHM) in (ms).

2.4. Oxidation model

The oxidation proceeds via a cubic rate law until the transition thickness (taken to be 2 μm) is accumulated [8,16]. That is,

$$\frac{ds}{dt} = \left(A/s^2\right) exp\left[-Q_1/RT_1\right], \tag{25}$$

after transition, the oxidation proceeds according to a linear rate law; that is,

$$\frac{ds}{dt}\left[C_o + U(M\Phi)^p\right] exp\left(-Q_2/RT_1\right), \tag{26}$$

where:

ds/dt = oxidation rate (μm/day),
T_1 = metal-oxidation interface temperature, °K,
ϕ = fast neutron flux (E>1 Mev), n/cm^2.s,
A = 6.3×10^9 μm^3/day,
Q_1 = 32,289 cals/mol,
C_o = 8.04×10^7 μm/day,
M = 1.91×10^{-15} cm^2.s/n,
P = 0.24,
Q_2 = 27,354 cals/mol,
U = 2.38×10^8 μm/day, and
R = 1.98 cals/mol/ °K

The oxidation layer thickness increments were calculated and added to the previous total thickness to obtain the end-of-step total thickness. The integration of the pre-transition equation was done without the feedback between oxide layer thickness and oxide-metal temperature. The oxide–layer thickness is converted to weight gain, and garzarolli et al.'s (1980) approximate integral solution is used and applied again over the current time step. The solution has the form [16]:

$$\Delta w_{i-1} = \Delta w_i + \frac{RT_o^2\lambda}{\gamma Qq''} ln\left[1 - \frac{\gamma Qq''}{RT_o^2\lambda} k_o\, exp\left(\frac{-Q}{RT_o}\right) exp\left(\frac{\gamma Qq''\Delta w_i}{RT_o^2\lambda}\right)\left(t_{i=1} - t_i\right)\right], \tag{27}$$

where:

i, $i+1$ is the refer to previous and current time step,
k_o is the rate constant, g/(cm^2-d) ,
T_o is the oxide-to-water interface temperature, °K,
λ is the oxide thermal conductivity, W/cm.°K
$\Delta\omega$ is the weight gain, g/cm^2,
q'' is the heat flux,W/cm^2, and
Q is the activation energy, cal/mol.

$$\gamma = 1.56[Zr]/2[O]\rho_{Zr} \quad (cm^3/g), \tag{28}$$

where Zr and O is the atomic weights of Zr and O in consistent units and ρ_{Zr} is the density of Zircaloy-4

2.5. FGR model at fast transient heating

In this section, the release of fission products from a reactor core during RIA was simulated. The release calculations require the values of the initial inventory of the species to be released from the core during the simulation, and current core temperature at each time step. Table 1 lists the amount of some significant fission products at the time of the accident (zero cooling time) for burn-up fuel of 10 & 45 MWd/Kg.U, respectively. Modeling of the release rates of fission products from the fuel is based on the models used in NUREG-0772 [1]. The amount M of the fission product remaining in the fuel after a time t is determined by:

$$\frac{dM}{dt} = -KM \ . \tag{29}$$

Where K is the fractional release rate, which can be defined as:

$$K = -(1/t)ln(1-F), \tag{30}$$

where F is defined as the fractional release of the fission product, so the F is equal to 1-M/M_o, where M_o is the initial amount of the fission product present, eq. (30) can be written as:

$$F = 1 - exp(-Kt), \tag{31}$$

the actual release for each fission product at each time step is the product of the fractional

release for that time step multiplied by the initial local inventory of each individual radionuclide. The initial inventory itself is a function of fuel burnup and operating temperature history. The fractional release rate coefficient is [17]:

$$K = K_o \exp(-Q/RT), \quad (32)$$

where:
K_o is the intercept value of K,
Q is the activation energy,
R is the universal gas constant = 8.314 J/mol K, and
T is the temperature in ^oK.

The corresponding values of K_o and Q for each fission product are presented in table 1.

3. Results and discussion

This section provides the simulation results of reactor neutronics, thermal-hydraulics and fission product release under RIA conditions. Some important safety parameters, such as power production, average fuel temperature, as well as release rate and fractional release of FPs are discussed. The transient response of the PWR depends on some initial plant conditions. Table 2 provides the input data, which are essential to initiate transient calculations. Emphasis must be given to a reasonable time step control in order to achieve physical meaningful results. Several time step criteria are employed and the minimum time step is selected. The RIA analysis is initiated by the ejection of a rod within 0.15 s and it is assumed that, during this time a reactivity of 1000 pcm is injected. The emergency scram is assumed to occur after 2s.

Figs. 1,2 show the relative concentration of the delayed neutron precursor concentration; C/C_o, and the neutron density; n/n_o, for an instantaneous jump of the reactivity: ρ = 0.95β, where the calculations are done using the finite-difference and Runge-Kutta methods. Comparison between the Runge-Kutta and finite-difference methods for the realistic control rod ejection is shown in fig. 3. A reactivity of ρ = 1000 pcm (~ 1.5 β or 1.5 \$) was injected within 0.15s. The other parameters were ρ_f = -0.32×10^{-4} and ρ_m = -1.286×10^{-4}. Fig. 3 shows the relative power (P/P_o) versus the transient time as calculated by the point kinetic model. The decay power is taken into account in the core power calculations. It is seen from figs. 1-3 that there is a good agreement between the finite-difference technique and Runge-Kutta method.

Fuel and clad temperature profiles during the transient as a function of time is indicated in fig. 4. This figure shows the surface and centerline fuel temperatures, as well as the inner and outer clad temperatures. The initial centerline and surface temperatures are 1362 ^oK and 960 ^oK, respectively.

The average and maximum fuel enthalpy difference versus time during RIA has been calculated using the average and centerline fuel temperatures. The calculated fuel enthalpy for reactivity insertion of ~ 1.5 \$ is shown in fig. 5. The enthalpies just before the pulse power generation were ~ 285 and 207 J/g UO_2 for the maximum and the average enthalpies, respectively. Pulse irradiation experiments with burned UO_2 fuel rods (25 MWd/kg U) showed that the failure of fuel rod occurs at enthalpy values much lower than the failure threshold value for fresh UO_2 fuel rods [18]. The burnup effect, i.e., decrease in cladding ductility by oxidation, hydrogen absorption and neutron irradiation as well as the accumulation of fission gas in fuel pellet, is considered to cause the rod failure at

Table 1
Inventory of some fine fission products

Element	*FP* mass (*g*) at average *Bu*=10 GWD/ton *U*	*FP* mass (*g*) at average Bu = 45 GWD/ton *U*	*Q* (3)	K_o (3)
Xe	1.624^{E}3	7.543E3	226	2E5
Kr	1.329E2	4.791E2	242	3.2E5
Cs	8.594^{E}2	3.735E3	292	8.1E5
I	7.308^{E}1	3.415E2	335	1.9E7
Te	1.389^{E}2	6.924E2	230	2.8E3

Table 2
PWR model parameter and initial conditions

Gross thermal power (MWt)	3411
Total fuel mass (ton)	101 UO_2
Enrichment (%)	3.2
Fraction of heat generation in fuel	0.951
Fuel specific heat (MJ/Kg.°K)	5.392E-3
Core flow rate (kg/s)	17400
Total coolant (kg)	3.06E16
Cross sectional area of the core (m^2)	3.37
Average coolant temperature (°K)	578
Centerline fuel temperature (°K)	1623
Fuel rod length (mm)	3810
Fuel rod diameter (mm)	9.7
% of power deposited in fuel rods	96
Fuel assemblies/core	241
Fuel rods/assembly	236
Clad thickness (mm)	0.813

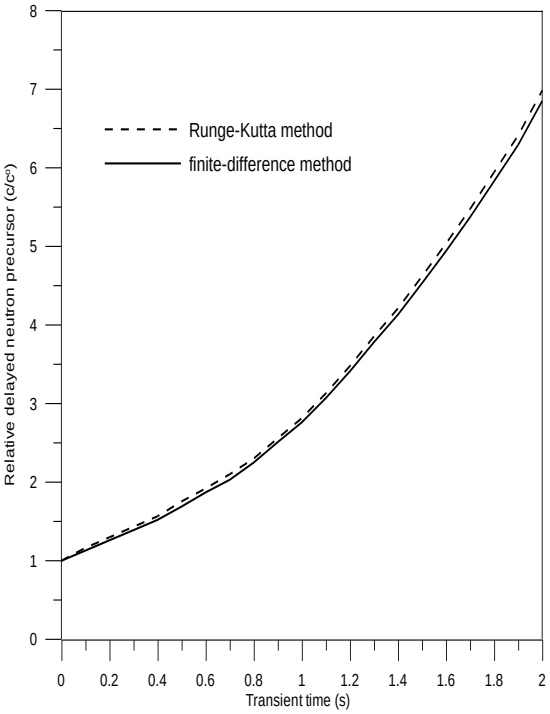

Fig. 1. Relative concentration of the delayed neutron precursor (c/c_o) for an instantaneous jump of the reactivity: ρ=0.95β, β = *0.0065.*

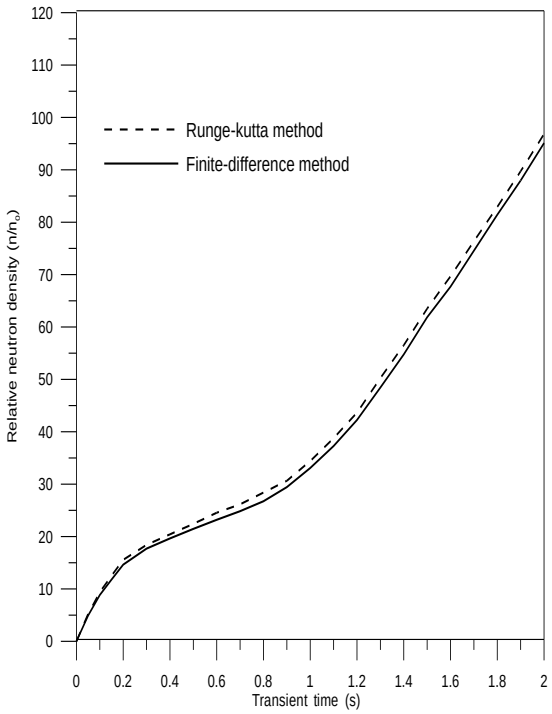

Fig. 2. Relative concentration of the neutron density (n/n_o) for an instaneous jump sof the reactivity: ρ=0.95β, β = *0.0065.*

relatively low energy depositions. The work of Sasajima et al. [18] has shown that the melting is the mechanism of fuel failure and the fuel enthalpy at failures is about 230 cal/g (963 J/g) for low burnup, and ~ 125 cal/g (523 J/g) for a burnup of 25 MWd/kgU. The experiments of Fuketa et al. [19] showed that the failure threshold enthalpy in the RIA at power lies in the range between 199 and 221 cal/g (832 J/g and 924 J/g). From the calculated results presented in fig. 5, we can conclude that no failures in the mode of cladding melting occur for the enthalpy range 335 J/g to 586 J/g.

Finally, by using the data shown in table 1, the release rate and the fractional release of

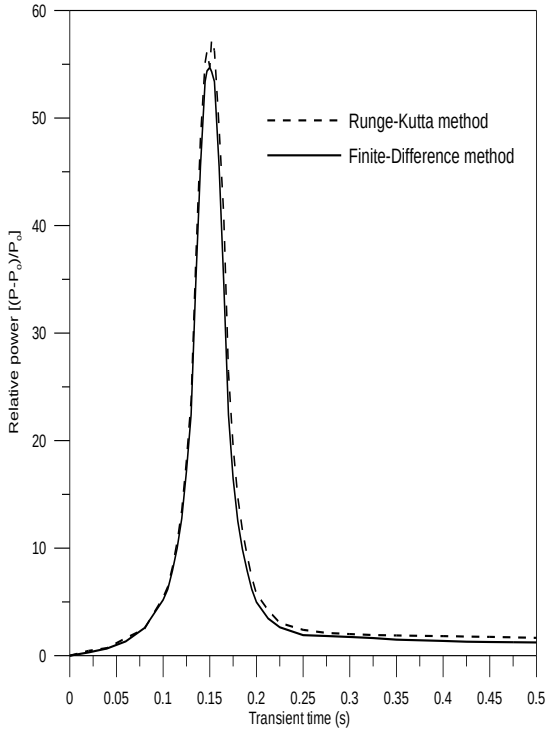

Fig. 3. Transient relative power pulse $[(p\text{-}P_o)/P_o]$ under RIA anlysis.

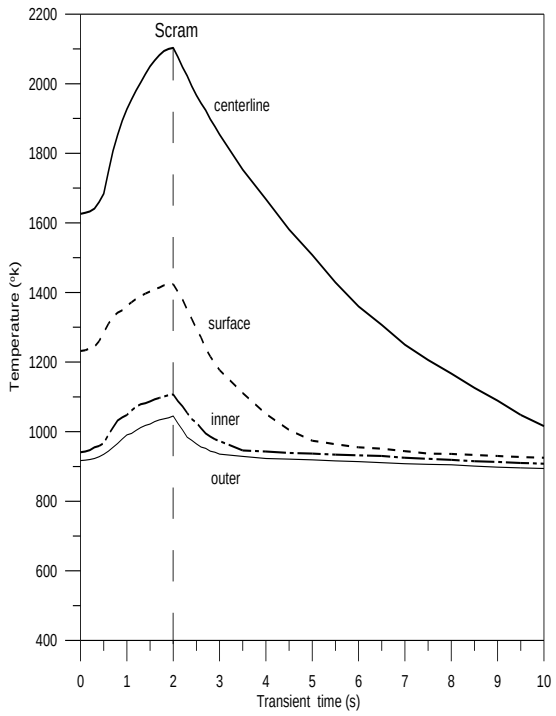

Fig. 4. Fuel and clad temperatures under RIA analysis-reactor scram is assumed to occue at 2 *s*.

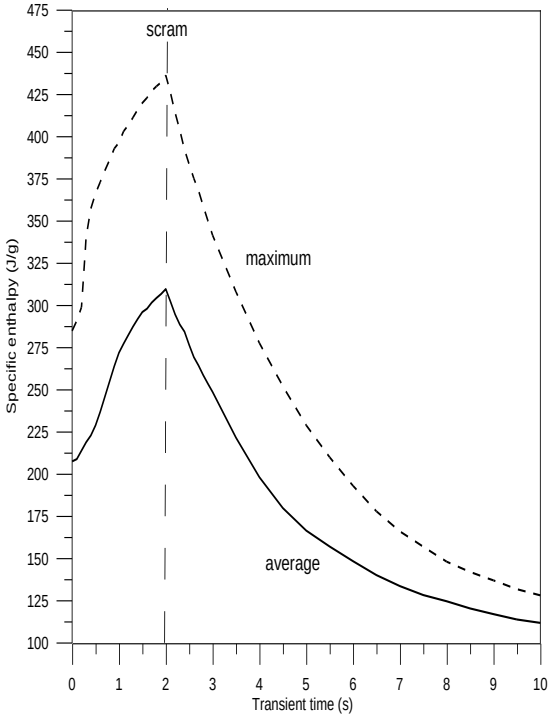

Fig. 5. Specific enthalpy versus transient time during RIA as calculated by the average centerline fuel temperatures.

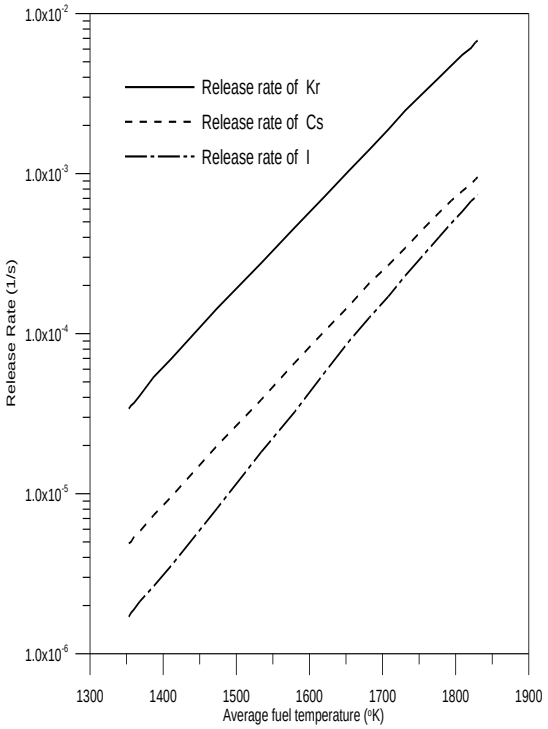

Fig. 6. Release rate of *Kr*, *Cs*, and as a function as a function of average fuel temperature.

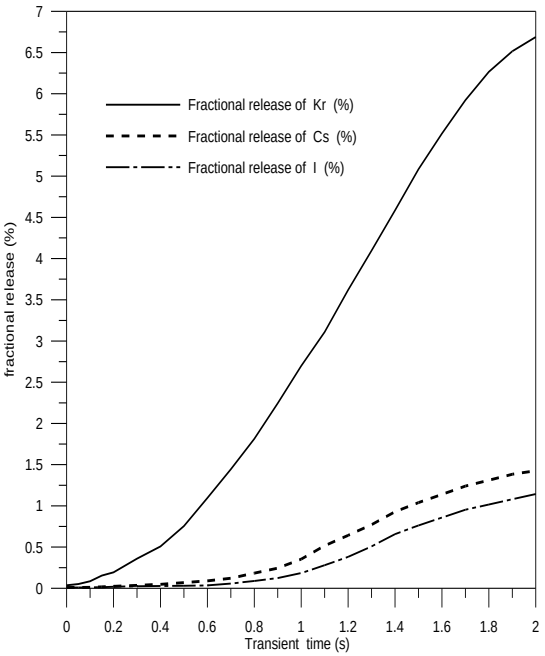

Fig. 7. Fractional release from the ruptured fuel to the cooling system of *Kr*, *Ca* and as a function of transient time.

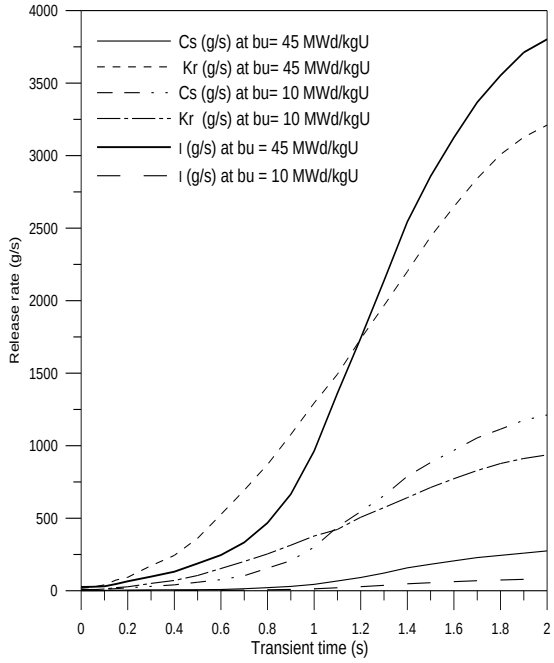

Fig. 8. Release rate (g/s) of *Kr*, *Cs* and /as a function of transient time.

some FPs (*Kr*, *Cs*, I) during the pulse irradiation were calculated as a function of the average fuel temperature and transient time, respectively. The results are shown in figs. 6-8. During the base irradiation, the FPs Release is very low (below ~ 0.1%), indicating that most of the fission products generated during the base irradiation remained in the fuel pellet. The FPs Release during the present pulse irradiation were estimated to be ~ 6.76 %, 1.43 % and 1.14 % for Kr, Cs, I, respectively. The free volume probably consists of pores and fission gas bubbles accumulated during base-irradiation and expanded by very sharp increase of temperature due to pulse irradiation. It is possible to consider that both thermal expansions of fuel matrix and bubbles resulting in additional volume increase of pellet could influence the deformation of cladding. The contact between pellet and cladding can occur when the swelling of pellet reaches at some threshold. From the results, it is concluded that the instantaneous increase in reactor power, such as power burst in RIA results in quick rise in the internal pressure of PWR fuel rods causing this peak in the FPs release.

4. Conclusions

The above results provide useful information to predict what happen in the reactor primary system during the first few seconds of the transient. It is concluded that, the estimation of the fuel temperature is to ascertain that the rate of heat removal from the reactor core is sufficiently high to prevent meltdown, and insured that the core integrity was maintained. From the above analysis, we can say that the PWR is safe during the first 5 s of the transient. Many models are involved in the program and their interconnections allow this program to take into account a large part of the physical phenomena occurring in the nuclear fuel during the first few seconds of the reactivity-initiated accident. The UO_2 fuel of LWR under RIA has been studied by simulating power burst of ~ 1.5 $. The emergency scram is assumed to occur after 2

s. As can be seen above, the transient is so long before the scram occurred. These results demonstrated that no failures of fuel rods occurred under the pulse irradiation of peak fuel enthalpy ranging from 335 J/g to 586 J/g in the mode of cladding melting. Significant FGR up to 6.76 % is estimated for Kr and 1.43 % and 1.14 % for *Cs* and I, respectively. However, the calculation will become more accurate when more extensive calibration is made based on RIA experimental data. Also, the model must be extended and improved for proper description of high burnup fuel behavior.

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Received November 21, 2004
Accepted March 17, 2005