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POWER EXCURSION ACCIDENT ANALYSIS OF RESEARCH WATER REACTOR

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A three-dimensional neutronic code POWEX-K has been developed, and it has been coupled with the sub-channel thermal-hydraulic core analysis code SV based on the Single Mass Velocity Model. This forms the integrated neutronic/thermal hydraulics code system POWEX-K/SV for the accident analysis. The Training and Research Reactors at Budapest University of Technology and Economics (BME-Reactor) has been taken as a reference reactor. The cross-section generation procedure based on WIMS. The code uses an implicit difference approach for both the diffusion equations and thermal-hydraulics modules, with reactivity feedback effects due to coolant and fuel temperatures. The code system was applied to analyzing power excursion accidents initiated by ramp reactivity insertion of 1.2 $. The results show that the reactor is inherently safe in case of such accidents i.e. no core melt is expected even if the safety rods do not fall into the core.

Keywords: *Gauss-Seidel iteration; reactivity accident, single mass velocity model, thermalhydraulic.*

INTRODUCTION

For the safety analysis of nuclear power plants, complex calculation models are used. These models are mainly consist of two different modules which account for the basic physical phenomena taking place in the plant: a neutronic module describing the neutron balance in the reactor core, and a thermal-hydraulic module simulating the heat transfer from the fuel to the coolant, the various evaporation and condensation processes which take place in the reactor core and in the condenser systems. The coupling between these two modules depends on the numerical schemes applied to integrate the whole set of equations constituting the model of the plant. These kinds of nuclear power plant models are too complicated for being applied to research reactors. The later require their own models.

Since, in general, the power level of the reactor is determined by the amount of reactivity added and rate of reactivity addition, all over power accidents are in essence, reactivity accidents. Nevertheless, it has seemed convenient to subdivide such accidents, and the first subdivision in this rather arbitrary classification has been called Reactivity

Excursions. This is meant to imply rather additions of reactivity resulting in rapid, usually non-oscillatory increase of reactor power.

The subject of the present article is to describe a model developed for simulating such accidents for the Training and Research Reactor of the Budapest University of Technology and Economics, (BME-Reactor in the following).

The training reactor, being located in the downtown of Budapest, has to satisfy special safety requirements, namely, it has to be inherently safe. It is required by the safety authority that the safety report of the reactor proved that such an accident could not lead to core damage and release of radioactive material. The former safety report written in 1996 used a combination of point reactor kinetic code REMEG and module HEATING of the SCALE 4.1 program system [1, 2]. This study was accepted at that time but the excess reactivity may not be increased until the study is repeated based on a more detailed and mathematically better founded computer model.

Due to the regular irradiation experiments performed at a power of 100 kW the excess reactivity needs to be increased to a value what the reactor had in 1980, i.e. near 1.2 $ in order to restore the original flexibility of the reactor operation.

The reactor can get permission for achieving this only if we are able to prove that even a prompt supercritical state of 1.2 $ can not result in a reactivity accident leading to core damage. It is trivial that it is not possible to check this statement experimentally at this reactor. Thus, we need a computer model simulating this accident.

In this article, the physical and mathematical models, features and status of both code POWEX-K and SV are described.

REACTOR DESCRIPTION

The BME-Reactor became critical in 1971. It is a tank type reactor, which is located on the site of the University. It is designed to be compact and safe and it is used mainly for neutron activation analysis, production of short-lived radioisotopes and for education and training. Further details can be found in references [2, 3]. A summary of the design and thermal-hydraulic parameters of the BME-Reactor core is given in Table 1.

Table 1: Design parameters of the BME-Reactor core.

Item	Value
Reactor type	Pool type
Reactor power level (kW, thermal)	100
Fuel type	EK-10
Enrichment (% in U-235)	10
Cladding	Aluminum
Moderator and coolant	H_2O
Reflector	Graphite+H_2O
Fuel radius (mm)	3.8
Thickness of the clad (mm)	1.2
Coolant flow rate (m^3/h)	5.8
Coolant inlet temperature ($^\circ$C)	20
Coolant inlet pressure (bar)	1.57
Total number of pins	369

1. POWEX-K/SV Code Models

1.1. Reactor Physics Model (POWEX-K)

The BME-Reactor dynamics under transient conditions have been modeled using the neutron diffusion equations [4]:

$$\mathbf{V}^{-1}\frac{\partial \mathbf{\Phi}(\mathbf{r},t)}{\partial t} = \nabla\big(\mathbf{D}(\mathbf{r})\nabla\mathbf{\Phi}(\mathbf{r},t)\big) - \mathbf{A}\mathbf{\Phi}(\mathbf{r},t) + (1-\beta)\boldsymbol{\chi}\mathbf{F}^{\mathrm{T}}(\mathbf{r})\mathbf{\Phi}(\mathbf{r},t) + \sum_{m=1}^{M}\boldsymbol{\chi}_m\lambda_m C_m(\mathbf{r},t) \qquad (1)$$

$$\frac{\partial C_m(\mathbf{r},t)}{\partial t} = \beta_m\mathbf{F}^{\mathrm{T}}(\mathbf{r})\mathbf{\Phi}(\mathbf{r},t) - \lambda_m C_m(\mathbf{r},t), \qquad m = 1,2,...,M \qquad (2)$$

where:

$\mathbf{r}$ = spatial coordinate
t= time (sec)
m= delayed neutron family index
M= total number of delayed neutron families
G= total number of energy groups
$\mathbf{\Phi}(\mathbf{r},t)$ = vector of space and time dependent neutron fluxes [n/cm^2/sec]:

$$\mathbf{\Phi}(\mathbf{r},t) = \begin{bmatrix} \phi_1(\mathbf{r},t) \\ \vdots \\ \phi_G(\mathbf{r},t) \end{bmatrix} \qquad (3.a)$$

$C_m(\mathbf{r},t)$= space and time dependent delayed neutron precursor concentration for delayed neutron family m (atom/cm^3)

$\mathbf{V}^{-1}$= diagonal matrix of inverse neutron velocities averaged for the energy groups (sec/cm):

$$\left[\mathbf{V}^{-1}\right]_{gg} = \frac{1}{v_g} \qquad (3.b)$$

$\mathbf{D}$= diagonal matrix of the neutron diffusion coefficients (cm):

$$\left[\mathbf{D}\right]_{gg} = D_g \qquad (3.c)$$

$\mathbf{A}$= total cross section plus slowing down matrix (cm^{-1}):

$$\left[\mathbf{A}\right]_{g,g'} = \begin{cases} \Sigma_g^{\mathrm{R}} + \Sigma_g^{\mathrm{a}} & g' = g \\ -\Sigma_{g'\to g} & g' \neq g \end{cases} \qquad (3.d)$$

$\boldsymbol{\chi}$ = vector of the prompt neutron fission spectrum:

$$\left[\boldsymbol{\chi}\right]_g = \chi_g^{\mathrm{p}} \qquad (3.e)$$

$\mathbf{F}$ = vector of neutron production cross sections (cm^{-1}):

$$\left[\mathbf{F}\right]_g = \nu\Sigma_g^{\mathrm{f}} \qquad (3.f)$$

$\boldsymbol{\chi}_m$ = vector of the neutron spectrum for delayed neutron precursor family m:

$$\left[\chi_m\right]_g = \chi_{mg} \tag{3.g}$$

β_m = delayed neutron fraction for family m

β = total delayed neutron fraction, $\beta = \sum_{m=1}^{M} \beta_m$

λ_m = decay constant for delayed neutron precursor family m ($\sec^{-1}$).

Superscript "T" denotes matrix transpose.

The POWEX-K[1] code solves Equations (1) and (2) by using the backward finite difference approximation for the time derivatives. The spatial derivatives are approximated by the usual finite difference scheme for XYZ-geometry [4]. We have developed a new iterative scheme for solving the 3D time dependent few-group neutron diffusion equation which simplifies the treatment of the delayed neutron precursors. It overcomes the difficulties associated with the normal source iteration techniques [5]. Our iteration formula reads as follows:

$$\mathbf{\Phi}_{ijk,\lambda+1}(t_\ell) = \left[\frac{1}{\Delta t_\ell}\mathbf{V}_{ijk}^{-1} + \mathbf{E}_{ijk} + \mathbf{A}_{ijk} - (1-\beta)\boldsymbol{\chi}\mathbf{F}_{ijk}^{\mathrm{T}} - \sum_{m=1}^{M}\frac{\lambda_m \Delta t_\ell \beta_m}{1+\Delta t_\ell \lambda_m}\boldsymbol{\chi}_m \mathbf{F}_{ijk}^{\mathrm{T}}\right]^{-1} \bullet$$

$$\bullet \Big\{ \mathbf{R}_{ijk}\mathbf{\Phi}_{i+1,j,k,\lambda}(t_\ell) + \mathbf{C}_{ijk}\mathbf{\Phi}_{i,j+1,k,\lambda}(t_\ell) + \mathbf{S}_{ijk}\mathbf{\Phi}_{i,j,k+1,\lambda}(t_\ell) +$$

$$+ \mathbf{U}_{ijk}\mathbf{\Phi}_{i-1,j,k,\lambda+1}(t_\ell) + \mathbf{H}_{ijk}\mathbf{\Phi}_{i,j-1,k,\lambda+1}(t_\ell) + \mathbf{B}_{ijk}\mathbf{\Phi}_{i,j,k-1,\lambda+1}(t_\ell) +$$

$$+ \frac{1}{\Delta t_\ell}\mathbf{V}^{-1}\mathbf{\Phi}_{ijk}(t_{\ell-1}) + \sum_{m=1}^{M}\frac{\boldsymbol{\chi}_m \lambda_m}{1+\Delta t_\ell \lambda_m} C_{m,ijk}(t_{\ell-1}) \Big\} \tag{4}$$

where subscript λ is the iteration counter. *Gauss-Seidel* method has been applied. The convergence of our iteration scheme was studied [5] and it was shown that the iteration is always convergent for subcritical reactor states while, for supercritical states, it is convergent only if time step Δt_ℓ is sufficiently small. It was equally shown that the Gauss-Seidel method accelerates the iteration in case of our model; too, just as it does in case of the conventional source iteration schemes.

The method has been found unconditionally stable. Due to the backward scheme, the time step Δt_ℓ is limited only by the speed of change of the neutron fluxes. Thus, relatively large time steps may be used for slow processes. However, they should be of the order of milliseconds for power excursion accidents.

Fuel temperature, moderator density, fission product poisons, and fuel burn-up are examples of feedback mechanisms that influence reactor dynamics. Feedback mechanisms can be distinguished by the response time to change occurring in the system. Fuel temperature and moderator density have time constants in the order of seconds. They are

[1] POWer Excursion-Khaled.

hours for fission product poisons and weeks or months for fuel burn-up. Since the power excursion transient is very fast, we may assume that only the fuel and moderator temperatures (consequently the moderator density as well) are the significant feedback mechanisms. These feedbacks are taken into account by recalculating the few-group constants at each time step for each node.

1.2. Sub-Channel Thermal-Hydraulic Model (SV)

The heat transfer model in SV[2] inside the fuel region is based upon one-dimensional radial heat conduction. The conservation equations are written for one-dimensional axial homogeneous upward flow through the channel. On the other hand, constitutive equations as heat transfer coefficient, friction factor are used taking into account the geometry as well as the convection regime (forced and natural). In the following, we shall discuss the basic models and equations used in both the fuel model and hydrodynamic model of SV module.

1.2.1 Thermal Heat Conduction Model

In SV we solve the one-dimensional conductive heat transfer from fuel to the clad and the convective heat transfer from the clad to the coolant for nuclear fuel rods to compute the rod internal temperature distribution and the rod surface heat fluxes. The heat conduction equation inside the fuel element is written in one dimension [6]:

$$\rho_F C_{pF} \frac{\partial T_F}{\partial t} = \frac{1}{r}\frac{\partial}{\partial r}\left(r k_F \frac{\partial T_F}{\partial r} \right) + \dot{q}_v''' \tag{5}$$

while for the cladding with no internal heat generation the equation is written as:

$$\rho_C C_{pC} \frac{\partial T_C}{\partial t} = \frac{1}{r}\frac{\partial}{\partial r}\left(r k_C \frac{\partial T_C}{\partial r} \right) \tag{6}$$

For the EK-10 type fuel pellet, there is no actual gap between the fuel outside surface and the clad, hence the accompanied set of boundary conditions for equations (5) and (6) are given by:

$$\left.\frac{\partial T_F}{\partial r}\right|_{r=0} = 0 \text{ and, } q'' = -k_C \left.\frac{\partial T_C}{\partial r}\right|_{r=r_{co}} = h[T_{co} - T_m]$$

where q'' is the heat flux [W/m^2].

The heat conduction and energy conservation equation are coupled at the interface between the clad and the coolant. The formula for h will be given later, where:

$T_F(r,t)$ = Fuel pellet temperature [K] $T_m(z,t)$ =Moderator temperature [K]

$\rho_F C_{pF}(t)$ =Fuel pellet heat capacity [J/m^3 K] $T_C(r,t)$ = Clad temperature [K]

$\rho_C C_{pC}(t)$ =Clad heat capacity [J/m^3 K] $T_{co}(z,t)$ =Clad surface temperature [K]

$k_f(t)$ =Fuel pellet thermal conductivity [W/m K]

[2] Single Mass Velocity

$\dot{q}'''_v(t,r,z)$ =Heat source per unit volume [W/m^3]

$k_c(t)$ =Clad thermal conductivity [W/m K]

r_{co} =Clad outside radius [m].

The thermal conductivity in [W/m K] and heat capacity in [J/kg K] for both the EK-10 fuel material type and aluminum alloy clad has been taken as functions of fuel and clad temperatures respectively [2, 7].

The finite difference method is applied in radial direction and is used for time derivatives and radial space derivatives. By applying a first order backward finite difference formula to Equations (4) and (5) at time $t = t + \Delta t$ are then given by:

$$\left[-\frac{\alpha_{ba}\Delta t}{\Delta r^2}+\frac{\alpha\Delta t}{2(i-1)\Delta r^2}\right]T_{i-1}^{n+1}+\left[\frac{2\alpha_{ce}\Delta t}{\Delta r^2}+1\right]T_i^{n+1}+\left[-\frac{\alpha_{fo}\Delta t}{\Delta r^2}-\frac{\alpha\Delta t}{2(i-1)\Delta r^2}\right]T_{i+1}^{n+1}=T_i^n+\frac{\alpha\dot{q}'''_v(n,i,j)\,\Delta t}{k_i^n} \tag{7}$$

where *n* is the time index and *i* is the radial index, and

$$\alpha=\frac{k_i^n}{\rho C_i^n},\quad \alpha_{ba}=\frac{\left(k_i^n+k_{i-1}^n\right)}{2\rho C_i^n},\ \alpha_{ce}=\frac{\left(k_{i+1}^n+2k_i^n+k_{i-1}^n\right)}{4\rho C_i^n},\quad \alpha_{fo}=\frac{\left(k_{i+1}^n+k_i^n\right)}{2\rho C_i^n}$$

where $i=1,2,...,M$, $T_i^{n+1}=T_i(t+\Delta t)$; $T_i^n=T_i(t)$ and

$$\dot{q}'''_v(n,i,j)=\begin{cases}\mathrm{E_f}\sum_{g=1}^{4}\Sigma_{\mathrm{f}}\Phi^g(n,i,j)\ \ [\mathrm{W/m^3}], & i=1,...,N\\ 0, & i=N+1,M\end{cases} \tag{8}$$

where $\dot{q}'''_v(n,i,j)$ again represent the heat source per unit volume in the i^{th} fuel radial node of the j^{th} axial mesh.

Equation (8) represents the interface between the neutronic field and the fuel heat conduction process.

where:

$\mathrm{E_f}$ =Energy per fission [W sec/fission]

Σ_{f} =Macroscopic fission cross section [1/m]

Φ^g =Group flux [fission /m^{2}sec]

ΔZ =Axial subinterval [m]

M=The maximum radial fuel and clad subdivisions

N=The maximum radial fuel subdivisions.

Equation (8) represents the interface between the neutronic field and the heat generated by the fuel heat conduction process. The matrix equation system (7) can be solved easily using the THOMAS algorithm [8].

In the calculations of this article, the core was subdivided into 30 axial and 10 radial nodes per channel. Heat transfer in each fuel element is determined on the basis of one dimensional conduction solution up to 29 axial sections. The moderator in each axial section is not subdivided. For the heat transfer across the clad-coolant interface, local flow and temperature dependent correlations were used.

1.2.2 Hydrodynamic Model

In the Single Mass Velocity model (SV) [9, 10] the two-phase flow treated as homogeneous flow with no slip, one dimensional vertical, and incompressible. In this model, the sonic effects are neglected. Moreover, it is assumed that the change due to the fluid expansion is neglected.

In the SV code module, the coolant flow and the connective heat transfer problem was solved by employing the momentum integral model which solves the governing mass, momentum and energy conservation equations along with the pressure balance equation for the entire domain on interest at each time step. The SV model has been found valid especially for small cores and very low coolant velocity like occurring in the BME-Reactor. The SV code can compute the flow and enthalpy distributions in nuclear fuel rod bundles and core for both steady-state and transient conditions. The SV code is very flexible for modeling a reactor core and fuel assemblies.

The general mass, momentum, and energy conservation equations are used in the form proposed by KAZIMI [9]. In the single mass velocity model the mass, momentum and energy conservation equations are:

$$\frac{\partial G_m}{\partial z} = 0 \tag{9}$$

$$\frac{\partial G}{\partial t} = \frac{1}{L}\left(\Delta P - F\right) \tag{10}$$

where ΔP is the pressure drop and F is the friction term.

$$\rho_m \frac{\partial H_m}{\partial t} + G_m \frac{\partial H_m}{\partial z} = \frac{q'' P_h}{A_z} \tag{11}$$

where:

A_z =Channel flow area [m^2] z =Axial distance [m]

G =Mass flux [$kg/m^2 sec$] H =Enthalpy [J/kg]

ρ =Density [kg/m^3] P_h =Heated perimeter [m]

t=Time [sec].

The equations we need to solve are Equations (9), (10) and (11). The finite difference form of equation (11) is [10].

$$(H_m)_j^{n+1} = (H_m)_{j-1}^n - \frac{1-\alpha}{1+\alpha}\left[(H_m)_{j-1}^{n+1} - (H_m)_j^n\right] + \frac{(q'')_{j-\frac{1}{2}}^n P_h \cdot 2\Delta t}{A_z(\rho_m)_{j-1/2}^n(1+\alpha)} \quad (12)$$

where $\alpha = G^n \Delta t / (\rho_m)_{j-1/2}^n \Delta z$ and j is the spatial index.

All the above values are referred to new time step i.e., n+1. Equation (16) can be changed into:

$$(G_m)^{n+1} = (G_m)^n + \frac{\Delta t}{L}(\Delta P^n - F^n) \quad (13)$$

After we know the mass velocity, we can use equation (12) to calculate the enthalpy and density distribution. Then determine the F^n accordingly [10]:

When, for a given power distribution inside the core, the thermal-hydraulic module computes the average fuel, clad and coolant temperatures for every mesh point of the finite difference scheme, we are able to calculate the few-group constants of the 3D diffusion equation. These few-group constants can be obtained by an asymptotic slowing-down and thermalization code (WIMS-D4 -4D in our case) [11]. The thermal-hydraulic feedback is taken into account via the recalculation of the few-group constants.

The solution is completely determined by considering the closure relationships [7]; the initial and boundary conditions. The initial distributions of $G_m(t)$ and $\mathrm{H_m}(z,t)$ are assumed known from steady state solutions [9]. The heat flux $q''(z,t)$ in a reactor is dependent on the coolant and fuel thermal conditions. Hence, the specification of $q''(z,t)$ is obtained from the effects of the neutronic response and the transient heat conduction in the fuel. In the present article we shall assume the inlet and outlet pressures are specified.

Now, Equations (1), (2), (5), (6), (9), (10), (11) together with the state equation and constitutive relations [7] or their corresponding finite difference forms are defining completely our integrated model.

RESULTS AND DISCUSSION

For the calculations performed for BME-Reactor by using the model described in the previous sections, the input data [7] used is depending on data from reference [1]. The calculations simulating power excursion accident due to a sudden increase of the reactivity from critical to 1.2 $ is presented in the following section. Since, according to the sense, such reactivity cannot be realized in BME-Reactor, the presented calculated results can not be verified experimentally. In order to have some comparison, the same results are shown for the point kinetics REMEG code. The comparisons give an idea about the improvement brought about by the 3D model with respect to the point kinetics model.

1. Power Excursion at Prompt Supercriticality

The desirable excess reactivity of BME-Reactor is 1.2 $, thus, the power excursion was analyzed for the ramp insertion of this reactivity. In this case, we adjusted a reactor size for which

$$k_{\text{eff}} = 1 + 1.2 * 0.00786 = 1.009432$$

where 0.00786 is the effective delayed neuron fraction for BME-Reactor as calculated by the kinetic version of program GRACE [12]. The adjustment was achieved by adding some fuel rods to the reactor. This is corresponds to the reality because this excess reactivity will be realized in the same way.

Figure 1 shows POWEX-K/SV predictions for the total reactor power. For illustration, the results are compared with the predictions obtained by means of the REMEG code. The figure shows that, starting from 1 kW, the power rises following a rapid exponential trend, increasing to 10.1 MW at 3.27 sec then it decreases due to the temperature feedbacks. REMEG increases till 14.8 MW at 0.5 sec, and then it decreases. It is remarkable that the two curves deviate from each other very much, namely the 3D model predicts a power peak which almost two times higher. The shutdown systems would fall in the reactor within 0.5 sec, thus they could reduce the power rise. However, as mentioned in the introduction, we did not assume the operation of the safety rods in order to see the effects of the inherent feedback mechanisms clearly.

The total energy released during the transient is 15514.2 kW sec, while it is 10776.17 kW sec for REMEG. If we divide these numbers by the nominal power of BME-reactor, we get 2.5 min and 1.8 min, respectively. This means that the energy released corresponds to the energy released during 2.5 min of nominal operation. In this basis, we do not expect very high temperatures. Figure 2 shows our model predictions for the fuel temperature while Figure 3 shows the clad and coolant temperatures at a fuel rod located near the core center and in the axial position located at half length of the fuel pin. The fuel temperature goes up to 538 ^{o}C and to 716 ^{o}C according to POWEX-K/SV and REMEG, respectively, for 7.8 sec and 4.99 sec following the ramp reactivity insertion. The clad temperature increased to 110 ^{o}C according to POWEX-K/SV while only to 71.8 ^{o}C according to REMEG in 6.8 sec and 4.9 sec. These figures are much lower than the melting point of aluminum (which is 660.2 ^{o}C). The moderator temperature for both codes is still less than 21 ^{o}C. The calculations were done up to 19 seconds until all the heat generated due to the transient went into the moderator and fuel and clad temperatures were stabilized. As in case of the reactor power, the temperatures calculated using POWEX-K/SV model are higher than those calculated by REMEG.

Figure 4 shows the time dependence of the thermal flux during the 1.2 $ transient for three different points inside the core: near the core center, near the reflector and near a water gap. These three curves are roughly proportional to each other: their ratios are nearly constant in time. This indicates that time dependence and space dependence of the flux are nearly separable from each. Taking into account that this separability is the fundamental assumption of the point kinetic model, the great deviations between the results obtained by POWEX-K/SV and REMEG cannot be explained by the differences of the 3D and point kinetic descriptions. Consequently, the differences are mainly due to a better thermal-hydraulic description of the power excursion by POWEX-K/SV.

Figure 5 shows the development of the normalized mass flux i.e. $G_m(t)/G_m(0)$ of the hottest channel during the 1.2 $ transient. It can be seen in this figure that no axial dependence according to the SV model and the mass flux reaches to more than 14 times its initial value in almost 17 sec. This is a natural convection initiated by the transient. It is an important difference with respect to the REMEG calculation which does not take into account convection.

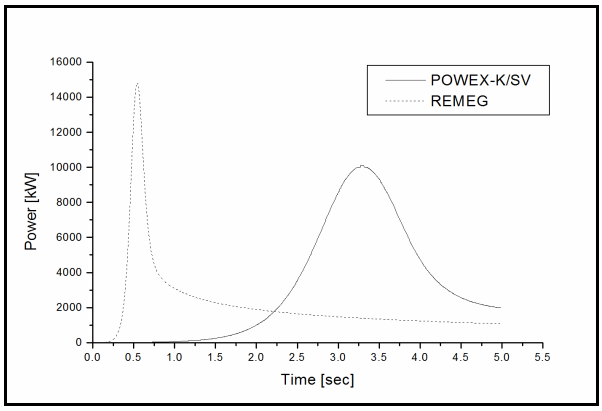

Figure 1: Time behavior of power for ramp reactivity of 1.2 $.

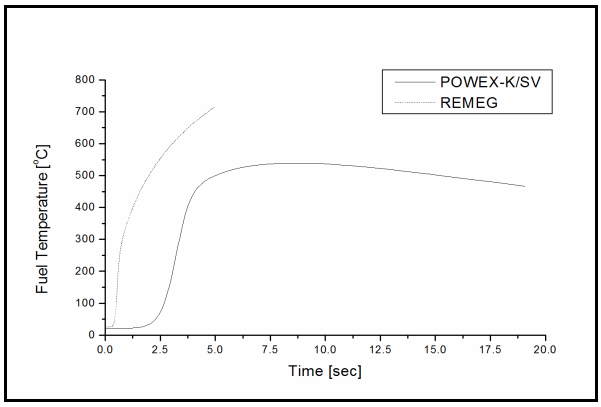

Figure 2: Time behavior of fuel temperature for ramp reactivity of 1.2 $.

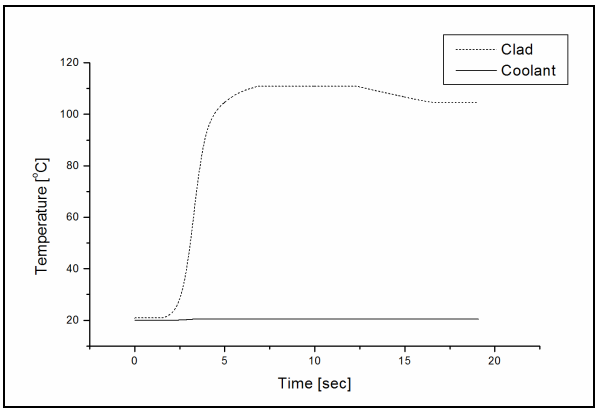

Figure 3: Time behavior of clad and coolant temperature three for ramp reactivity of 1.2 $.

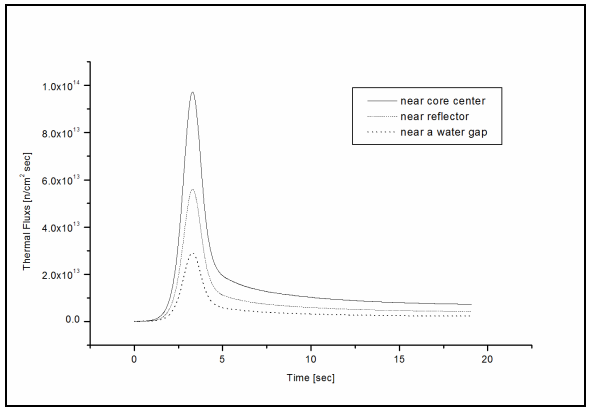

Figure 4: Thermal flux as a function of time in different points inside the core for 1.2 $.

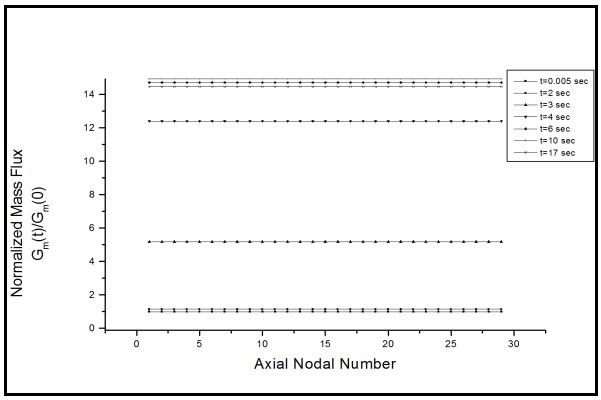

Figure 5: Normalized mass flux of hottest channel for 1.2 $

CONCLUSION

From the results presented in the previous section, we can draw the following conclusions:

- From the comparisons with the results of the point kinetic REMEG code, we may conclude that the 3D model predicts the reactivity accident more severe both as far as peak powers and temperatures are concerned. This is due to a better thermal-hydraulic description by POWEX-K/SV.
- Even for 1.2 $ reactivity insertion, we need not reckon with the melting of the clad and boiling of the moderator.
- It can be noticed form Figure 1. that the transient occurred much faster in REMEG than in POWEX-K/SV. Moreover, the temperatures are much less in our model than in REMEG. This is due to the much better 3D treatment of the problem.

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