Assessment of Error from the Use of Point Kinetics When Analyzing Transients in the Large Size Fast Reactor

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Abstract. "Point kinetics" approximation is widely used in the stage of reactor design justification primarily for analysis of transients and emergency modes. Analytical study of error caused by the use of point kinetics approximation was performed on test model of the large size fast reactor with "flat core" with the use of coordinated space dependent and point kinetics models within the framework of UNICO multi-physical code (3D neutronics in diffusion approximation + 3D thermalhydraulics). Three test transient problems were studied: change of coolant temperature at the inlet of reactor core diagrid (in one of four primary loops), safety rods drop at nominal power (example of fast running process), and unauthorized withdrawal of one control rod. It is shown that in case of unauthorized movement of one control rod the error of point kinetics estimation of the fuel element cladding temperature as compared to that from space dependent model can be as high as 100 ºC.

Key Words: fast reactor, sodium, safety, software, point kinetics error.

1. Introduction

"Point kinetics" approximation is widely used in the stage of reactor design justification primarily for analysis of transients and emergency modes. Point kinetics model is taken as a basis of the Russian codes DINROS, GRIF, and SOKRAT-BN used for fast reactor safety analysis. Its popularity is explained by its relative simplicity and physical transparency (the possibility to interpret the results in terms of reactivity effects and easily demonstrated verification). Approximation error for traditional fast reactor designs is sufficiently low.

Point kinetics approximation is primarily based on the postulate of the possibility to neglect spatial neutronics effects, or rather to neglect their possible change during the course of processes under study. In other words, form factor determining spatial distribution of neutronics parameters (reactivity effects and power density) is calculated in advance, as a rule, for the initial state, and then power behavior is determined by the amplitude factor.

On the other hand, there are some problems, in which local parameter changes are critical from the standpoint of reactor core reliability and safety analysis. For instance, fuel elements having maximum cladding temperature at a certain time can be found in the different fuel subassemblies during transient. Test model of 2,800 MW(th.) sodium cooled fast reactor has been considered. The reactor has flattened core (H/D = 0.2) of large diameter (4.2 m). The core consists of 1,408 hexagonal ducted subassemblies of various designs spaced in the triangle lattice with 185 mm pitch. In the central part of the core there are 432 subassemblies with U-Pu nitride fuel pellets of single enrichment and 31 subassemblies modeling control rods.

Analytical study of error caused by the use of point kinetics approximation was performed for test model of the large size fast reactor with “flat core” with the use of coordinated space dependent and point kinetics models within the framework of UNICO multi-physical code (3D neutronics in diffusion approximation + 3D thermalhydraulics). Three test transient problems were studied: change of coolant temperature at the inlet of reactor core diagrid (in one of four primary loops), safety rods drop at nominal power (example of fast running process) and unauthorized withdrawal of one control rod.

2. Mathematical models

For the purpose of correct comparison of space-dependent and point kinetics models, it is necessary to provide a maximum agreement in the data used in the models. In the space dependent approach the isotope composition (and the corresponding macroscopic cross-section) in each calculation zone is taken as an input data. Withdrawal or insertion of the control rod is simulated by changing related concentrations (and macroscopic cross-section constants) in the area of the rod.

As regards point kinetics, the rate of addition of the outer reactivity should be preset directly. In order to determine this parameter, a series of steady state calculations was made to find k_{eff} values for different positions of the moving rod. The similar approach was used for determination of temperature reactivity coefficients, the only difference being in varying temperature of corresponding material. In the course of mathematical simulation the main attention was paid, in particular, to the agreement of reactivity effect values obtained by point kinetics and space dependent kinetics approaches. Net reactivity determined by the point kinetics model is a sum of the following components:

$$R_T = R_{Na} + R_{St} + R_D + R_{out},$$

where

R_{Na} - sodium related reactivity component due to sodium temperature change;

R_{St} - steel related reactivity component due to steel temperature change;

R_D - fuel Doppler reactivity component;

R_{out} - outer reactivity added according to preset law due to control rods movement.

Temperature effects were evaluated using simple model, in which these effects were determined by the behavior of the average temperature of the core elements. For instance, reactivity increase due to the sodium temperature change was determined by the following relationship:

$$R_{Na}(t) = a_{Na}(T_{Na}(t) - T_{Na}(0)),$$

where $T_{Na}(t)$ and $T_{Na}(0)$ - current and initial values of average sodium temperature in the core, respectively; a_{Na} – corresponding reactivity coefficient.

Contribution to reactivity caused by steel temperature change can be presented by the similar relationship:

$$R_{St} = a_{St}(T_{St}(t) - T_{St}(0)),$$

where $T_{St}(t)$ and $T_{St}(0)$ - current and initial average steel temperatures in the core, respectively (including fuel element claddings and subassembly ducts); a_{St} – steel temperature related reactivity coefficient. Doppler effect is described in the same way using Doppler constant k_D and fuel temperatures $T_f(t)$ at the moment t and $T_f(0)$ at $t = 0$. Reactivity coefficients obtained by the MAG module and used for the point kinetics calculations are presented in Table 1.

TABLE 1. REACTIVITY COEFFICIENTS FOR REACTOR TEST MODEL

Parameters	Values
Reactivity coefficient due to sodium in core a_{Na}, (ΔK/K)/°C	-7.498E-7
Reactivity coefficient due to steel in core a_{St}, (ΔK/K)/°C	5.49e-7
Doppler constant k_D, ΔK/K	-6.14e-3

Neutronics module MAG [1-3] is based on the similarly-named code designed for solving steady state and transient neutron transport equations in 1D, 2D, and 3D geometry. Space dependent dynamics in subassembly-to-subassembly presentation was described by the transient diffusion group equations in 3D Δ-Z geometry. The parameterized low-group nuclear data library was used. Fuel temperature, sodium density and steel temperature were used as parameters. Reactivity effects were evaluated in the series of steady state calculations with varied approximation parameters. A series of steady state calculations was also carried out to determine the component related to addition of outer reactivity R_{out}.

Thermalhydraulic module SACTA has been designed on the basis of similarly-named code designed earlier [4-6]. In SACTA module, reactor core model is an ensemble of the fuel subassemblies submerged into the inter-subassembly sodium, and actually a dual problem is solved on heat transfer between sodium flowing inside fuel subassemblies and that flowing in the inter-subassembly space. In this module, simplified 3D thermal hydraulics model is used for calculation of sodium temperature pattern within the subassembly, as well as 3D thermal conductivity model describing temperature distribution in the impermeable cylindrical elements, namely: fuel elements, shield subassemblies and control rods.

Agreement of spatial distribution of neutrons and thermalhyrdraulic parameters at $t = 0$ was achieved by simple iterations.

3. Results of test problems solution and comparative analysis

Problem 1. Asymmetric change of coolant temperature at the inlet of the core diagrid (in one out of four primary loops)

In the initial state, reactor is operating on rated power. Sodium temperature at the core inlet is 410 ºC. Stepwise change of the core inlet sodium temperature by 100 ºC was assumed as an initial disturbance. To simplfy analyses we suppose that if in one inlet nozzle sodium temperature increased by 100 ºC, then the temperature in all other inlet nozzles decreased by the value just sufficient for maintaining constant average sodium temperature at the core inlet (410 ºC).

In spite of the constant average sodium temperature at the core inlet, the azimuthal heterogeneity of temperature pattern in this area would cause change of the average temperatures throughout the core. The transient turned out to be rather long, this obviously being caused primarily by significant thermal inertia of inter-subassembly sodium (its slow heating).

Net reactivity value in Problem 1 is positive, so reactor power decreases by 3-5 %. So, the absolute value of the effect itself is small, however the nature of dynamics forecasted by two different approximations, differs even qualitatively. Reactor power spike just after coolant temperature disturbance at the core inlet evaluated by spatial kinetics model is much larger than that calculated by point kinetics approximation. There is also difference in asymptotic power values. To make it more demonstrative, Table 2 gives values of maximum fuel temperatures ($T_f^{\max SD}$ for space-dependent approach and $T_f^{\max PK}$ for point kinetics), maximum cladding temperatures ($T_c^{\max SD}$ for space-dependent approach and $T_c^{\max PK}$ for point kinetics) calculated and differences $\delta f = T_f^{\max SD} - T_f^{\max PK}$ and $\delta c = T_c^{\max SD} - T_c^{\max PK}$ at different time points.

TABLE 2. MAXIMUM FUEL AND CLADDING TEMPERATURES CALCULATED BY THE DIFFERENT APPROXIMATIONS AT DIFFERENT TIME POINTS, AND DIFFERENCES BETWEEN SPACE-DEPENDENT AND POINT KINETICS VALUES.

Time, s	Fuel			Cladding		
	$T_f^{\max SD}$	$T_f^{\max PK}$	δf	$T_c^{\max SD}$	$T_c^{\max PK}$	δc
5	1491.2	1491.2	0	605.6	605.6	0
50	1576.7	1572.9	+3.8	701.3	700.7	+0.6
200	1562.5	1569.6	-7.3	698.9	700.1	-1.2
300	1555.8	1568.8	-13.	697.8	700.0	-2.2

1000	1546.5	1567.2	-20.7	696.3	699.7	-3.6

In the early stage of the process, point kinetics approximation underestimates fuel and cladding temperatures, respectively, by 4 ºC and grade fraction. Later on, the difference between temperature values obtained by two approximations increases and its sign changes. After 1000 s point kinetics overestimates fuel and cladding temperatures by 20 ºC and 3-4 ºC, respectively.

Problem 2. Safety rods drop in the reactor operating on rated power. In the initial state, reactor is operating on rated power. Safety rods are in their upper position. At the moment τ = 2 s, all six safety rods are inserted into the core. Their total weight, according to calculation by MAG module, is –0.018ΔK/K. The rods are inserted at constant speed during time interval of 1 s. Integral neutron power is evaluated in point kinetics approximation for this initial event at the high accuracy. On the other hand, local power decrease is observed in the fuel subassemblies situated in the vicinity of safety rods (15 % decrease by the 50-th second). Calculation made in point kinetics does not show this effect, and power of all fuel subassemblies, including that adjacent to the rod, decreases simultaneously. The effect of approximation on the accuracy of calculation of the core temperatures for this initial event should be acknowledged insignificant so we don't presents detailed results for this problem..

Problem 3 Unauthorized withdrawal of one control rod. In the initial state, reactor is operating on rated power. At the moment τ = 2 s, one control rod starts moving upwards at the speed of 10 mm/s. At this speed, control rod would be withdrawn to its upper position during 53 s. According to the calculation made by MAG module, the total weight of withdrawn control rod is – 0.31%Δk/k (0.7858βeff). Since the rod starts moving from about its medium position, then the net inserted reactivity value is 0.228%Δk/k (0.578 βeff).

Control rod withdrawal and related positive reactivity addition cause increase of the core temperature and onset of the new level of reactor power and core temperature. The inserted reactivity is compensated by temperature effects. Difference between the results obtained in point and spatial kinetics approximations is significant even for the integral parameters. For instance, in case of point kinetics approximation the steady state integral neutron power is underestimated by 10 % with its total increase of 65 %. The differences of local fuel and cladding temperatures are even larger. Figs. 1 and 2 show behavior of fuel and cladding temperatures in two fuel subassemblies located in the vicinity of moving control rod (assembly M-630), although at the different distances from the rod. Fuel subassembly M=631 borders on the withdrawn control rod, and fuel subassembly M=632 is located at the distance of one row from the rod. Steady state temperatures of fuel and cladding and their variations when calculated using different approximations are presented in Table 3. Point kinetics calculation for the fuel subassembly adjacent to the control rod underestimates fuel temperature at the top of the core by 341 °C (the effect is even stronger in the core central area), and cladding temperature is underestimated by 101 °C. For the fuel subassembly separated by one row from the control rod the effect decreases, although it remains significant, namely: 236 °C and 69 °C, respectively, for fuel and cladding.

TABLE 3. STEADY STATE FUEL AND CLADDING TEMPERATURES AND THEIR DEVIATIONS DEPENDING ON THE TYPE OF APPROXIMATION USED FOR NEUTRON KINETICS MODELING

Elements	Fuel subassembly (M=631)			Fuel subassembly (M=632)		
	Spatial kinetics	Point kinetics	δ	Spatial kinetics	Point kinetics	δ
Cladding	735	634	101	726	657	69
Fuel	1,364	1,023	341	1,345	1,109	236

Thus, in the analysis of accident caused by unauthorized withdrawal of control rods the error in calculation of power density change can result in the underestimation of key parameters by point kinetics approximation, upon which the consequences of the accident are estimated.

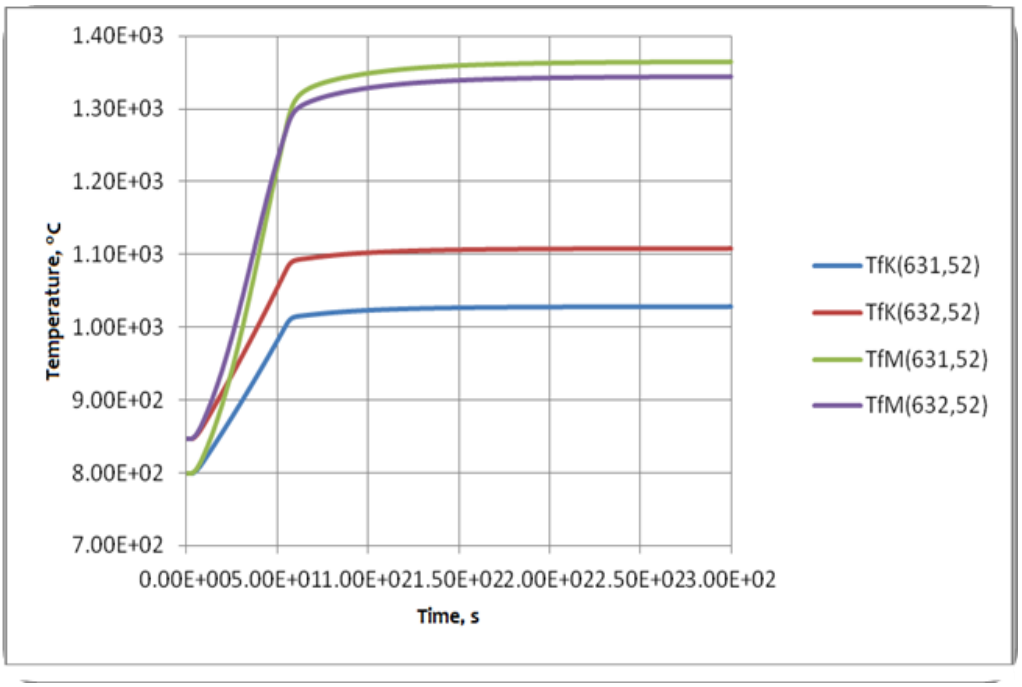

Fig. 1 Dynamics of local fuel temperature at the core top in the subassemblies adjacent to the withdrawn control rod, TfK - fuel temperature with point kinetics, TfM -fuel temperature with MAG(Space-dependent kinetics)

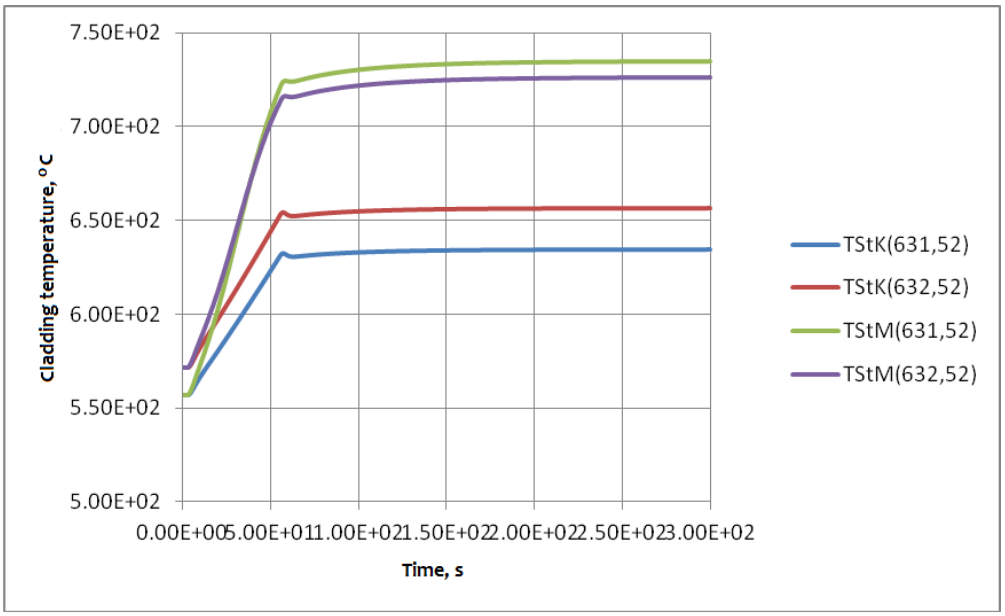

Fig. 2 Dynamics of local cladding temperature at the core top in the subassemblies adjacent to the withdrawn control rod, TStK - steel temperature with point kinetics, TStM -steel temperature with MAG(Space-dependent kinetics)

4. Conclusions

Point kinetics model is relatively simple. Consequently, calculation of any process by this model needs much less computer working time, this being an evident advantage of this approach.

However, the use of point kinetics model requires the following input data:

a) all calculated temperature reactivity coefficients;

b) determined initial subassembly-to-subassembly power density distribution in the core (form factor).

Obtaining this data requires many neutronics calculations and their amount will be even larger when it is supposed to use so called “scenario” approach in the hope of more precise modeling of reactivity effects. Also, recall that in case of “scenario” approach (postulate key events and use point kinetics from one key event to another one) implementation, on the one hand, additional efforts will be required for designing approximation functions describing all components of the reactivity effect, and, on the other hand, as is well-known, it is impossible to avoid potential simulation errors even within the framework of “scenario” approach, since it is impossible to foresee in advance all possible conditions emerging in the course of accident. The latter is particularly true with regard to severe accidents with potential partial dry-out of the core and relocation of molten fuel and steel. So, the apparent ease of use of point kinetics masks complex of time-consuming preliminary calculations and undoubted inaccuracy of description of some accident processes. If multi-physical code including spatial kinetics neutronics module is used, then the above drawbacks no more exist.

Value of the error appearing in case of point kinetics depends on specific problem, however, it is always significant with regard to the value of the effect under study. For instance, even in the problem of asymmetric coolant temperature disturbance at the core inlet, although

absolute value of the effect related to neutron power change is only 3.5 %, however, in case of point kinetics approximation, this effect is underestimated actually by a factor of 1.5. In case of accident caused by unauthorized withdrawal of the control rod the errors of calculation of the key parameters are significant (both relative and absolute values), and point kinetics approximation underestimates not conservatively the values of core elements temperatures. The error of the fuel element cladding temperature calculation by point kinetics approximation can be as high as 100 ºC as compared to that from the space dependent kinetics.

Thus, in general it can be stated that point kinetics approximation should be used with an abundance of caution even in case of analysis of transients and accidents resulting in no core meltdown and also if the strict requirements are imposed on speed-of-response, for instance, in the simplified simulator type models. Multi-physical code including spatial kinetics module is more convenient to use for the analysis of accidents from practical point of view, and it is certainly capable of providing higher accuracy in process modeling. The advantages of such approach manifest even more obviously in the modeling of severe accidents assuming core meltdown and relocation of molten material.

5. References

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