SENSITIVITY ANALYSIS AND VERIFICATION OF FUEL ROD MODEL USED IN COUPLED NEUTRONIC AND THERMAL-HYDRAULIC CODES

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ABSTRACT

Realistic modeling of heat transfer in fuel rods is very important for coupled computer codes, because temperature-coefficient feedback is responsible for interfaces between neutronic and thermal-hydraulic parameters of VVER reactor core.
Key parameters for fuel temperature calculations such as fuel-clad gap and fuel thermal conduction greatly vary with burnup. The problem is in that they are modeled in modern codes or measured in un-loaded fuel with heavy uncertainty.
The paper analyzes the sensitivity of core thermal and neutronic calculations to the above-mentioned uncertainty. The study is carried out on the basis of SAPFIR_95&RC program package. Heat transfer in fuel is described with a numerical model which considers the following processes in fuel:

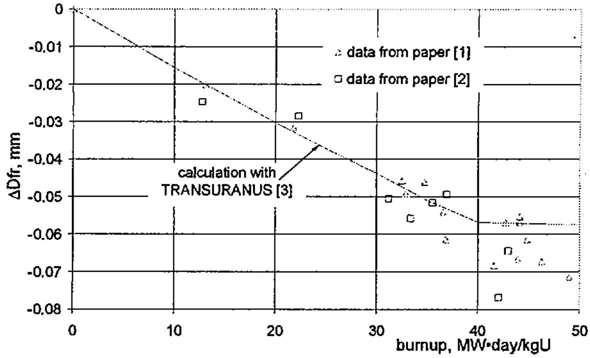
- thermal expansion;
- fuel swelling and cladding creep;
- degradation of fuel thermal conduction and change of gap conductance.

The calculation of the fuel effective temperature accounts for asymmetric disposition of pellets in rods. Experiments on excitation of free axial xenon oscillations highly sensitive to fuel temperature-coefficient feedback were used for model verification.

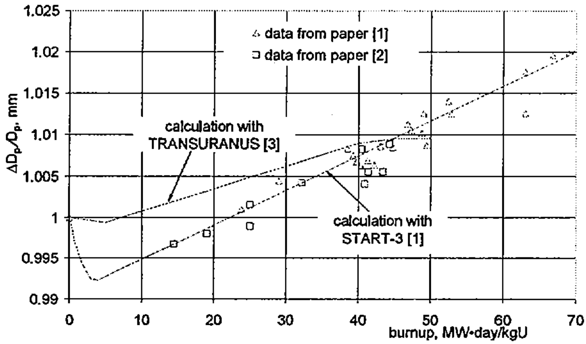
1. FUEL ROD GEOMETRY VARIATION WITH BURNUP

Experimental data on cladding creep and fuel swelling under fuel irradiation conditions have been taken from papers [1] and [2] (See Figure 1).

Figure 1, b) presents also two fuel swelling curves calculated by Medvedev et al. [2] with START-3 program and by Passazh et al. [3] with TRANSURANUS program.



a) average variation of fuel rod diameter



b) relative variation of fuel pellet diameter

Fig. 1 – Variation of fuel clad and pellet diameter under burnup conditions in VVER-1000 reactor

Calculation with START-3 program is in good agreement with experimental data on fuel swelling under high-burnup conditions (Figure 1,b), but there is no experimental evidence of when and to what extent fuel is maximally compressed at the beginning of burnup occurring as a result of fuel sintering. The minimum of the curve calculated with START-3 program is approx. 1.5-1.6% of initial cold fuel volume and falls on burnup fraction of 3 to 4 MW·day/kgU. Calculation with TRANSURANUS program confirms the assumption of some authors regarding the existence of an incubation period in fuel swelling at the beginning of burnup.

Scatter in experimental data relative to their approximating curve with burnup more than 14 MW·day/kgU is about 8%.

When describing heat transfer in the fuel-clad gap considering the above error, we can neglect some effects whose contributions to gap conductance under normal operating conditions are smaller than the error arising from gap width uncertainty. These effects are thermal resistances caused by fuel and clad roughness, incomplete temperature adaptation, fuel-clad contact and conductance caused by nuclear radiation.

Fission gas accumulation in the fuel-clad gap under normal operating conditions has little impact on fuel temperature. According to paper [2], sufficient amount of fission gas accumulates beneath fuel clad after burnup of 45 MW·day/kgU. Under these burnup conditions, the gap «collapses» as a result of thermal expansion and fuel swelling and practically does not affect fuel temperature evaluation. Presence of fission gas in the gap corrects fuel temperature by no more than 0.5 to 1.5°C for the whole fuel burnup range.

Numerical models for fuel and clad deformation under burnup conditions were developed to study the influence of parameters uncertainty on fuel temperature calculations and core neutronic characteristics. Figure 2 shows modeled curves for fuel rods used in the reactor core of the Rostov NPP.

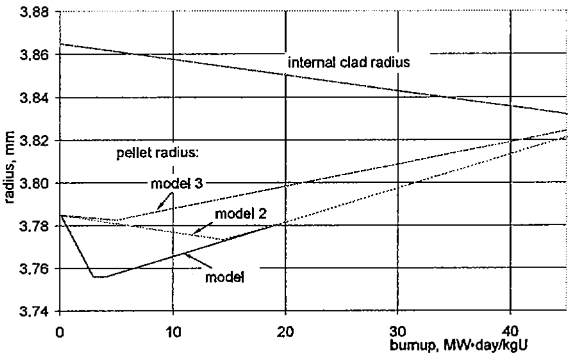


Fig. 2 – Numerical models for pellet and clad deformation under burnup conditions. Fuel rod designed for power unit #1, the Rostov NPP

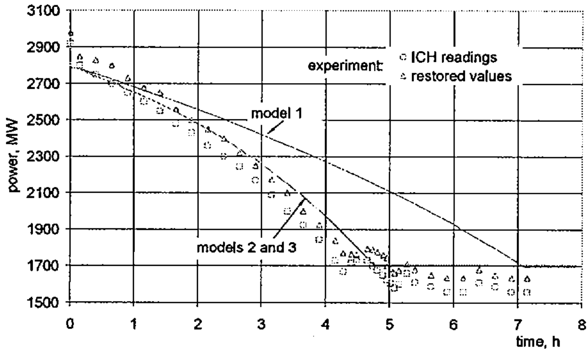
Actually, they simulate variations of gap width under burnup conditions. Indirect estimations are used to verify the models.

2. MODELING OF EXPERIMENTS ON CORE STABILITY UNDER XENON OSCILLATIONS

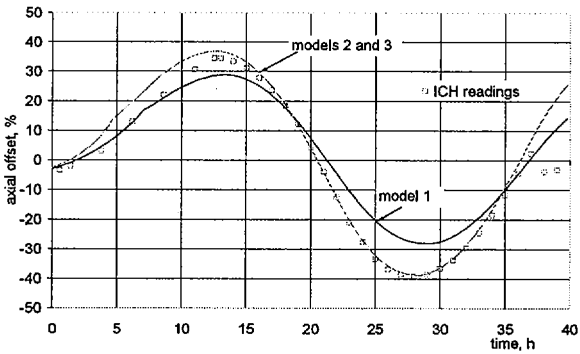
The dependence of average fuel temperature on power determines the value of reactivity power effect that is the key parameter in providing core stability under integral power fluctuations. The error of 10% in reactivity power effect evaluation has no influence on the results of steady-state neutronic calculations. But modeling of transients connected with integral power change or in-core power density redistribution demonstrates that even small difference in reactivity power effect evaluation can have much impact on core behavior with time. High sensitiveness to negative temperature-coefficient feedback is observed, for instance, in modeling the experiments with excited axial xenon oscillations.

The experiment of this kind was carried out during the first fuel cycle on power unit #1 at the Rostov NPP with burnup of 80 EFD (approx. 4 MW·day/kgU), which corresponds to the burnup where maximum fuel compression is achieved in model 1 (see Figure 2), models 2 and 3 have similar values for fuel diameter.

The experiment consisted in reducing core power by approx. 5% by injection of some boric acid into the coolant of primary circuit. Xenon buildup started that resulted in further power reduction at a growing rate with the automatic power regulator being off. Other core parameters did not change. When the power reached the value of approx. 1700 MW, it was controlled at this level by changing boric acid concentration in coolant. Free xenon oscillations were developed at constant core parameters values. Modeling results are given in Figure 3.



a) 1st stage — post-perturbation spontaneous power reduction



b) 2nd stage — free axial xenon oscillations

Fig. 3 (a,b) – Modeling of axial xenon oscillations excited by small power reduction Rostov NPP, power unit #1, 80 EFD.

In this experiment power reduction rate (see Figure 3, a) is affected by xenon accumulation rate and fuel temperature decrease. Curves in Figure 3 demonstrate that models 2 and 3 are in good agreement with experimental data. Model 1 underestimates both power reduction rate at the 1st stage of experiment and the amplitude of axial xenon oscillations.

Figure 4 presents the results of modeling power reduction in the experiment on iodine well depth measurements on power unit #1 at the Rostov NPP with burnup of 56 EFD.

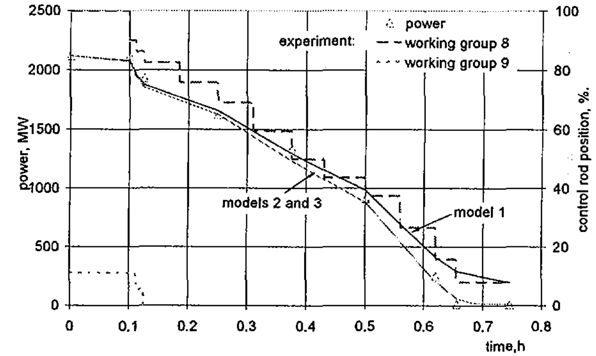
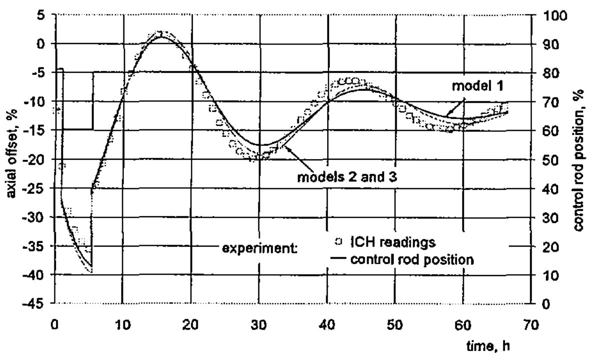


Fig. 4 – Power reduction in the experiment on iodine well depth measurements.
Rostov NPP, power unit #1, 56 EFD.

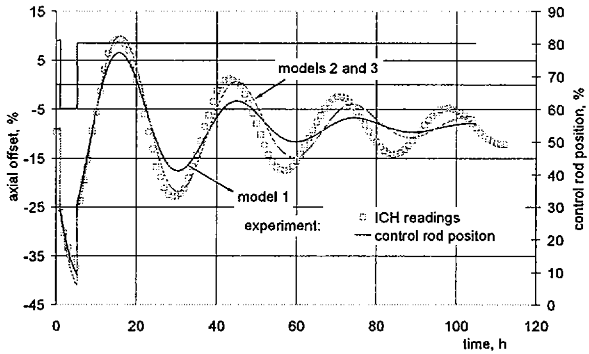
In this experiment, power decreases at a faster rate as a result of control rod insertion, and poisoning effects are not as significant as in the previous case. Negative fuel temperature feedback slows the process down.

To make sure that the best agreement of models 2 and 3 with experimental results is not connected with the error in calculating the reactivity coefficient with respect to fuel temperature changes, additional experiments were simulated where this coefficient was not determinative.

In these experiments axial xenon oscillations were excited by short-time immersion of control rods at constant integral core power. The first two experiments were performed during the first fuel cycle on the power unit #1 at the Rostov NPP. These experiments have different burnup fractions - 30 and 60 EFD, which corresponds to approx. 1.5 and 3 MW·day/kgU, respectively. Modeling results are presented in Figure 5.



a) burnup of 30 EFD



b) burnup of 60 EFD

Fig. 5 (a,b) – Axial offset changes after short-time immersion of control rods.
Rostov NPP, power unit #1.

As in the experiment with integral power reduction, using of models 2 and 3 for fuel rod geometry variation results in the lower fuel temperature and, consequently, the larger amplitude of xenon oscillations, which is in better agreement with experimental results.

The third experiment of this kind with burnup of 175 EFD was carried out at the Zaporozhye NPP (see Figure 6). This experiment was not quite clean, because power and control rod position varied constantly, which made the experiment modeling more complicated.

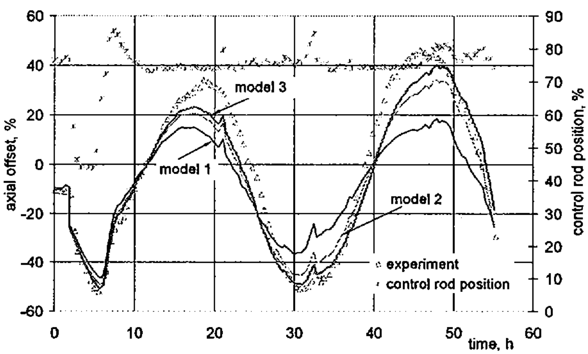


Fig. 6 – Axial offset changes after short-time immersion of control rods.
The Zaporozhye NPP, power unit #1, 175 EFD.

With burnup of 175 EFD (approx. 8 MW·day/kgU), model 2 and model 3 come apart: fuel pellet diameter decreases in model 2, while in model 3 it is practically the same as that of a new fuel rod. Difference in average fuel temperature between model 2 and model 3 is 15°C. Even this small temperature difference affects modeling results, which demonstrates high sensitivity of these experiments to temperature-coefficient feedback.

The results of modeled integral experiments do not confirm strong compression of fuel at the beginning of burnup and in all cases speak in favor of model 3 with the lowest average fuel temperature in fuel assembly at least for burnup fraction of less than 10 MW· day/kgU.

3. INFLUENCE OF AXIALLY MISALIGNED FUEL PELLETS ON THE RESULTS OF TRANSIENTS MODELING

Asymmetric disposition of fuel pellets in rods is one more factor increasing the uncertainty of model results.

Constant data sets used in neutronic programs are obtained for fuel temperature constant across the pellet section. Actually, fuel temperature increases sharply from the outside toward the fuel rod center. To calculate the reactivity power coefficient correctly, it is necessary to use effective fuel temperature value that is calculated by averaging temperature distribution over fuel rod radius with weight function [4]:

$$\varpi(r) = 1/\sqrt{T(r)}.$$

This weight function enhances the contribution of outer fuel layers having lower temperature to effective temperature evaluation. In a fuel rod with axially aligned pellets the effective temperature is 6-7 °C lower than the volume-averaged temperature. If fuel pellets are misaligned from fuel rod axis, average fuel temperature does not practically change, while the effective temperature decreases as a result of minimal temperature decrease in outer fuel layer. The maximum effect is obtained if all pellets are maximally misaligned from fuel rod axis and touch the clad inner wall. In this case, the effective fuel temperature at nominal power is 30 °C lower than the average temperature.

Figure 7 shows models for spontaneous core power reduction caused by core power perturbation using different methods for calculation of effective fuel temperature.

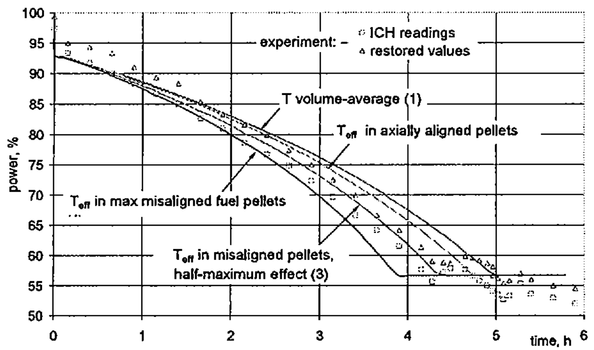


Fig. 7 –Modeling of core power reduction using different methods for calculation of effective fuel temperature

The upper curve (1) is obtained using volume-averaged fuel temperature as an effective temperature. The curve (2) is calculated for the effective temperature in a fuel rod with axially aligned pellets. The lower limit of uncertainty presents the curve (4) obtained using the effective fuel temperature in a rod with all pellets maximally misaligned from fuel rod axis.

The best model results are obtained when the effective fuel temperature is calculated using the half-maximum effect of pellet misalignment. Asymmetrical fuel disposition effect disappears under burnup conditions.

4. ANALYSIS OF INFLUENCE OF FUEL THERMAL CONDUCTION UNCERTAINTY ON MODELING OF CORE PARAMETERS

Apart from the fact that thermal conduction has a direct impact on evaluation of maximum fuel temperature and, as a consequence, is involved in reactor core safety evaluation, its value depending on temperature and burnup fraction affects core behavior under transient conditions.

Some thermal conduction data [2, 5, 6, 7] is presented in Figure 8. There is one-and-a-half-fold difference between thermal conduction estimates in operating temperature range.

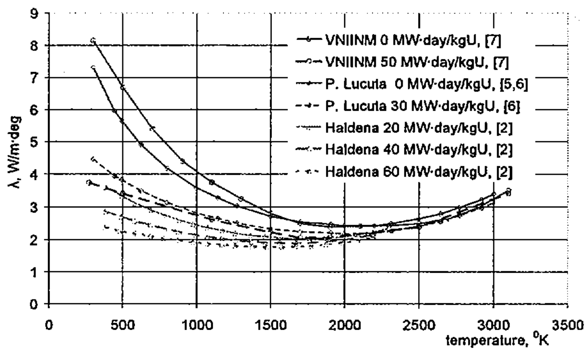


Fig. 8 – Fuel thermal conduction

SAPFIR_95&RC program package is based on the curve taken from VNIINM data base [7]. This curve was used to carry out verification calculations under program package certification procedure.

Figure 9 shows modeling of power reduction (1st stage of experiment on axial xenon oscillations excited by small power reduction with burnup of 80 EFD) with thermal conduction data taken from papers of P. Lucuta [5,6]. In this case average fuel temperature in the core increases by 25-30°C.

Using of lower thermal conduction results in lower power reduction rate, which can be explained by the fact that lower changes in power are required to compensate reactivity loss caused by xenon poisoning, because:

$$\Delta N \sim \lambda \Delta\rho_{xe} / (\partial\rho / \partial T_U) \sim \lambda \,.$$

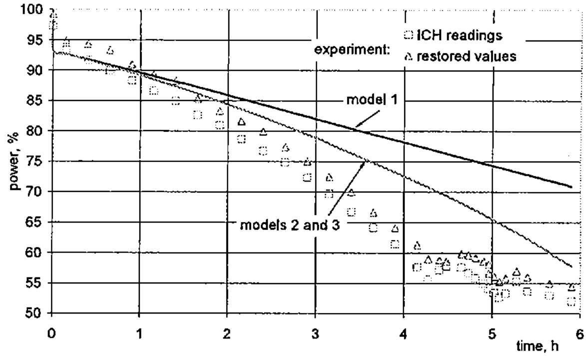


Fig. 9 – Modeling of core power reduction. Calculation with thermal conduction values from papers of P. Lucuta [5,6]

Consequently, the lower is fuel thermal conduction, the higher is the value of negative power effect and the more stable is the core behavior under integral power fluctuations.

Comparison of calculation results shows that neutronic model done with RC program is in better agreement with experimental results, if fuel thermal conduction values from VNIINM data base are used for calculations. Fuel thermal conduction data taken from papers [5,6] overrates core stability under integral power fluctuations. Using of thermal conduction data from paper [2] increases fuel temperature by 50-60^{0}C and cause a significally larger discrepancy between calculated and experimental data.

5. CONCLUSIONS

Based on comparison of calculated and experimental results, the following conclusions can be drawn:

— fuel rod model 3 based on calculations of paper [3] is recommended to consider clad and fuel deformation at the beginning of fuel burnup of 0 to 14 MW·day/kgU;

— fuel thermal conduction data from VNIINM data base [7] are recommended for use in coupled computer codes;

— in neutronic calculations it is recommended to use the fuel effective temperature corrected for axial misalignment of pellets in fuel rods.

— the uncertainty of fuel rod model parameters can be shortened using indirect estimations based on full-scale modeling of integral experiments carried out at the operating power units. The experiments of this kind can help to make numerical models less conservative.

To verify the fuel rod model at higher burnup, it is necessary to extend experimental base used for fuel rod model verification by performing analogous experiments under high burnup conditions at the end of fuel cycles.

ABBREVIATIONS:

NPP	—	nuclear power plant
ICH	—	ionization chamber
EFD	—	effective full-power day
D_p	—	pellet diameter
D_{fr}	—	outer fuel rod diameter
N	—	power, W
T	—	temperature
λ	—	thermal conduction W/m·deg
ρ	—	reactivity

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