Code Package to Analyze Behaviour of the WWER Fuel Rods in Normal Regimes of Operation: TOPRA-s Code

A. Scheglov, V. Proselkov

RRC Kurchatov Institute, Moscow, Russian Federation

Presented is the brief description of the code package (WWER Division INR RRC KI) intended for the analysis of characteristics of the WWER fuel rods, or their sections. Included are:

- TOPRA-1 and TOPRA-2 codes to analyze full-scale fuel rods;
- MRZ and MKK codes to analyze separate sections of fuel rods in *r-z* and *r-φ* geometry.

Presented is the description of the TOPRA-s code intended for the express-analysis of thermophysical parameters of cross-sections of the WWER fuel rods: thermal conductivity through the fuel-cladding gap and temperature distribution along the fuel radius when reactor is in normal operation.

Presented are the code validation results against the data obtained in the experimental programs: SOFIT-1 (1.1-1.4) and IFA-503.1. Comparison of calculation results obtained by TOPRA-s code and by TRANSURANUS code (version for WWER reactors) is also presented here.

Results obtained in the process of verification indicate the possibility of using the method and the code (TOPRA-s) for the simplified engineering analysis of thermophysical parameters of the WWER fuel rods. TOPRA-s code is certified by GAN of RF. TOPRA-s code has been included into KASKAD neutron code package to generate data on the fuel temperature.

VVERD INR RRC KI is the scientific adviser of the WWER cores. Therefore, it is necessary to have complete set of computer codes for the analysis (or assessment) of the behaviour of all the elements of the WWER core including fuel rods. Although it is ARSRIIM who is the general designer of fuel rods, VVERD performs additional justification of the working ability of fuel rods and analyzes their physical parameters in the process of operation. It is very important, for example, to know such parameters as fuel temperature in the fuel rods to calculate neutron and physical characteristics of the core.

A number of computer codes are used to analyze WWER fuel rod parameters in the regimes of normal operation. Among these codes are TOPRA, TOPRA-2, MKK, MRZ, and TOPRA-s.

1. Codes to Analyze Full-Scale Fuel Rods

TOPRA-1 code (PIN-mod2 [1]) is intended for modelling of the behaviour of thermophysical parameters of fuel rods and fuel-and gadolinium rods of the power and research reactors of the WWER type, as well as for justification of their working ability in the quasi-steady-state regimes. The code is developed on the basis of PIN-micro [2] (1991 version) code. Within several years some of the PIN-micro models were totally replaced, some models were modified. The obtained version of the code was named PIN-mod1, and later – PIN-mod2.

PIN-mod2 code was verified against experimental results obtained in MR, MIR and HBWR reactors (in the framework of SOFIT, FGR-2 and FUMEX experimental programs), and against the results of comparison of the post-reactor examination of the WWER-440 and –1000 fuel rods with the calculation results (see, for example, [1,3]). In 1997 the name of PIN-mod2 code was changed for TOPRA, as this code had practically nothing in common with the initial PIN-micro code. The code accounts for the changes in operating conditions, structural and technological parameters of fuel rods, as well as the fundamental processes that take place during operation and influence the fuel rod behaviour.

It is important to note that TOPRA-1 code is intended for the assessment of the fuel rod working ability from the standpoint of meeting thermophysical criteria, only. In TOPRA-1 code analyses of the cladding diameter changes, cladding elongation due to mechanical interaction with fuel, fuel-cladding contact pressure, radiation densifying and swelling of fuel can only be performed with the simplified semi-empirical dependencies.

TOPRA-1 (PIN-mod2) code was used to justify working ability (from the standpoint of meeting thermophysical criteria) of the WWER fuel rods for some fuel cycles, in particular:

- 4- and 5-year fuel cycles for Rovno NPP, Units 1 and 2;
- Transition to the 4-year fuel cycle of Kola NPP, Unit 3;
- Leaving of 12 fuel assemblies for the 6-th year of operation of Kola NPP, Unit 3 [4].

TOPRA-2 code was developed in 2000 by introducing into TOPRA-1 code of an additional block to model thermal mechanics of the fuel and cladding deformation. The block was developed by Prof. Tkachev, and Mr. Zheltukhin with participation of the authors of this report. The code is intended for modelling of thermophysical and mechanical (strength) characteristics (including the parameters of the stress-strain state, checking of the working ability criteria, etc.) of the fuel and fuel-and Gadolinium rods of the power and research reactors. It is also intended for justification of their working ability in the quasi-steady state regimes of operation. During the analysis the fuel rod power history (time dependencies of the linear power distribution, of the fast neutron flux, of the coolant pressure and temperature along the height) is divided into time steps. Automatic selection of the step size is performed in the beginning of every step with the use of data calculated for the previous step.

The fuel rod is analyzed in the so-called sesqui-dimensional approximation. It is divided along the height into axial zones combined by axial stresses and common composition and pressure of the gas media inside the cladding. Gas in the gas collector is accounted for.

Methods of the elasticity, plasticity and creeping theories [5,6] are used for the strength analysis. Cladding deformation is calculated according to the streaming theory. Anisotropy of zirconium tubes is accounted for. Modelling of the fuel deformation is performed in accordance with the modified aging theory. Coupled (via friction and pressure) non-linear boundary problem of the joint fuel and cladding deformation is solved. According to the analytical model of the “thick-wall cylinder” fuel and cladding are analyzed as separate elements.

The code is being verified currently, and the used properties of the fuel and cladding are being specified.

2. The Codes to Analyze Separate Sections of Fuel Rods

Usually the model of continuous coaxial cylinders is used in the codes calculating working ability characteristics of the full-scale fuel rods. Still geometrical and physical deviations from this model are typical of a real fuel rod. These deviations are due to:

- Use of pellet fuel;
- Facets and minor holes in the pellets;
- Cracking of fuel causing concentration of stresses in the cladding in the place, where it faces the crack – in the conditions of complete fuel-cladding contact.

And possibly due to:

- Buckles and ovality of the surfaces;
- Eccentric location of the fuel and cladding;
- Fuel pellet chippage and getting of the fuel chips into the fuel-cladding gap;
- Axial asymmetry of the volumetric heat generation in the fuel;
- Axial asymmetry of heat removal from the cladding outer surface;
- Presence of gaps between fuel pellets in a fuel rod. These gaps can be present in the initial fuel rods and can be formed in the process of operation due to additional fuel sintering after the fuel rod reaches the power level, or due to jamming of the fuel column with the distorted pellets.

Due to the impact of these deviations there exist differences in the fuel and cladding temperatures, in the local heat flux from the cladding surface and in the cladding stresses from the values calculated by the sesqui-dimensional codes. Cladding ovality impact can increase the local heat flux by up to 20% [7,8]. Cracks in the fuel can increase growth of the hoop stresses by 20% [7] on the cladding inner surface close to the place, where the crack faces the cladding under the conditions of the fast power growth and the fuel-cladding gap. It is important to account for the impact of the gap between fuel pellets (in the fuel rods with the gap and in the neighboring ones) onto the increase of the local heat flux from the cladding to the coolant. This increase is used to calculate one of the components of the engineering margin coefficient.

MKK and MRZ codes [7] were developed on the basis of the finite element method with the use of the three nodal finite elements. They are intended for 2-D analysis of the temperature field in the local sections of fuel rods. The codes also allow to calculate migration of points in the analyzed region caused by thermal expansion, as well as stresses in the elements of this region in the framework of elastic problem (MKK code). In 1999-2001 the following was done to modernize the codes:

- New properties of the fuel, cladding and gas media were introduced;
- Consideration of dependencies of the relative energy deposition and burnup along the fuel radius versus average burnup in the pellet was introduced;
- Close-meshed dissection of the nodes was introduced.

At the same time the codes were verified. Verification was performed in the form of comparison of the calculated values:

1. Temperatures, migrations, and stresses (for MKK) with the results of analytical solutions and calculations by other codes for 1D task;
2. Temperatures with the results of analytical solutions of 2D tasks.

Verification results of the 1D and 2D temperature analysis indicate that the accuracy of calculations is better than $\pm 0.2\%$ in respect to the temperature difference in any point of the fuel rod and of the coolant temperature. Verification results indicate that the codes demonstrate good accuracy in the analysis of migrations and stresses.

It is assumed that the TOPRA output data will be used as the input data necessary for the analysis with these codes (geometrical parameters –

with consideration of the changes versus burnup, fuel density, composition and pressure of the gas media).

MKK code [7] is used to calculate the cross-section of fuel rods in (r-φ)-geometry. This code was used to analyze some characteristics of the working ability of fuel rods (of the WWER and thermoemission types).

MRZ code [7,9] is used to analyze local sections of fuel rods in (r-z)-geometry (for example, one and a half pellets with the adjoining cladding). This code was used to analyze the increase of the local heat flux from the fuel claddings in WWER due to the gaps between fuel pellets.

3. TOPRA-s Code

TOPRA-s code [10,11] is intended for the express analysis of thermophysical parameters of cross-sections in the WWER fuel rods during normal operation of the reactor:

- Thermal conductivity through the fuel-cladding gap or contact;
- Temperature distribution along the fuel radius, and the fuel average temperature.

The code accounts for the operating conditions of the fuel rod, its structural and technological parameters. Solution of a thermophysical task in 1D r-geometry is laid into the basis of modelling to develop the code. This solution is based on analytical thermophysical correlations and physical models describing fundamental processes that take place in a fuel rod during its operation and influence the temperature field in it. The reviewed fuel cross-section is divided into 21 zones along the radius. Both the gap and the cladding are represented as one zone. The code models are first of all based on the data of post-reactor examinations of experimental and standard fuel rods, as well as on experimental studies of the material properties, on experimental and design-theoretical studies of the processes inside the fuel, on the results obtained with TOPRA-1 and TBC-M codes [12].

Depending on the input data reactor type (WWER-440 or 1000), average for the fuel rod burnup and linear power, operating conditions of the reviewed cross-section (burnup, linear power, and the coolant temperature) this code determines the state of the reviewed cross-section, including thickness of oxide layers on the cladding surfaces, current values of the fuel and cladding radii, fuel-cladding gap size, coefficient value of thermal conductivity of gas media in the gap and calculates cladding surface temperatures, value of the fuel-cladding gap thermal conductivity, and temperature distribution along the fuel radius accounting for distribution of the relative power density and burnup along the fuel radius.

It is shown in [10], that for the WWER fuel rods it is possible to significantly simplify the models describing the diverse processes taking place in the fuel rod during the in-reactor operation, without any significant precision worsening of the calculated fuel temperature field. At that, it is possible to avoid the use of the preceding (up to the calculated time point) power history and to consider each of the fuel rod cross-sections separately using the rod average linear power and burnup only. Before all, this concept can be applied to "typical" fuel rods, i.e. to rods operating at working conditions, representative for the most of the fuel rods of a particular reactor design at steady state (base) operation conditions. On the base of this specific approach, the TOPRS-s code has been developed.

The code was developed so that the results calculated by it could be interpreted as the most typical for fuel rods of the respective burnup and linear power.

3.1. Possibility to Simplify the Models

Calculation results of the temperature field in a fuel rod cross-section, except for the coolant temperature and linear power, cladding and fuel thermal conductivity coefficients, are mainly influenced by:

- Fuel-cladding gap size (or the contact pressure);
- Thermal conductivity of gas media in the gap;
- Change of the burnup and relative power density along the fuel radius;
- Other factors (changes of: density of fuel, oxide layers on the surface, fuel open porosity, etc.). Using the data of post-reactor examinations of the WWER fuel rods (e.g. [4,13-15]) it is possible to consider influence of these factors on the fuel temperature as the function of the cross-section burnup with good accuracy.

1. For the WWER fuel rods there exists a lot of information explaining behaviour of the fuel-cladding gap. This allows to predict its change in the process of burnup with good confidence. Up to the burnup value of 35-40 MWd/kgU change of the fuel-cladding gap is mainly caused by the creep, radiation growth and thermal expansion of the cladding, cracking, volumetric changes and thermal expansion of the fuel. Decrease of the gap size due to creeping and radiation growth of the cladding, radiation additional sintering and swelling of the fuel can be predicted on the basis of the data of post-reactor examinations [13-15]. Dependencies for the relocation (decrease of the gap size due to the fuel cracking) can be obtained by comparison of the calculated results with the experimental data. For the burnup higher than ~45 MWd/kgU it is possible to assume that there is no fuel-cladding gap under operating conditions.
2. Thermal conductivity of the gas media in the gap depends on its composition, pressure

and temperature. Gas composition inside the cladding depends on the amount and composition of the initial filling gas and on the fission gas that has released from the fuel. Based on the data of post-reactor examinations and computer analysis it is possible to use for the WWER fuel rods simple enough dependencies of the fission gas release versus burnup. For example, for the WWER-440 fuel rods it is possible to assume that fission gas release is equal to 0.45-0.7% if the average burnup in the fuel rod is 30-45 MWd/kgU. Growth of the fission gas release takes place at a higher level of burnup. In the WWER-1000 fuel rods fission gas release is somewhat higher, but it is possible to predict it [13]: up to the burnup of ~45 MWd/kgU it is 0.5-0.8%, at the burnup of 50 MWd/kgU it reaches ~3%. Value of the gas pressure inside the fuel rod cladding as the function of its average burnup and linear power can be inputted by using for example TOPRA-1 calculation results.

3. Changes of the burnup and relative power density along the fuel radius of full-scale fuel rods are analyzed with neutronic codes or blocks. These changes do not as a rule depend on a fuel power density.

Therefore, for the WWER fuel rods consideration of the changes (as the function of burnup and of the reactor type WWER-440 or 1000) of the major parameters influencing the temperature field in them can be performed with the help of simple dependencies. At that, it is possible to obtain dependencies not using the previous history of the fuel rod operation.

3.2. Explanation of a Possibility to Review a Separate Fuel Rod Cross-section

In the codes analyzing full-scale fuel rods links between axial zones are established via the composition and pressure of the gas media inside the cladding, via the stress values, and elongation of the cladding length. If thermal conductivity of the gas media in the gap is accounted for correctly (its temperature and composition), then in order to calculate temperature field in the fuel rod cross-section it is enough to review this cross-section, only.

3.3. Code Verification

The TOPRA-s validation [10] has been carried out through comparison of the calculated results with experimental data for instrumented fuel rods irradiated in RNC KI, MR (SOFIT-1 [16]), GNC RF NIIAR MIR (experiment FGR-2), HBWR (FUMEX1, rod №1 and the WWER type fuel IFA-503.1&2). The comparison has been based on the fuel centerline temperatures at the cross-sections with thermocouples. In Figures 1 and 2 [10], the results of all compared data (as differences between the maximum fuel and the coolant temperatures) of the SOFIT-1 and IFA-503.1 rods, during the entire period of in-reactor operation, are summarized. The drawn straight lines can be seen in these two figures, as well as

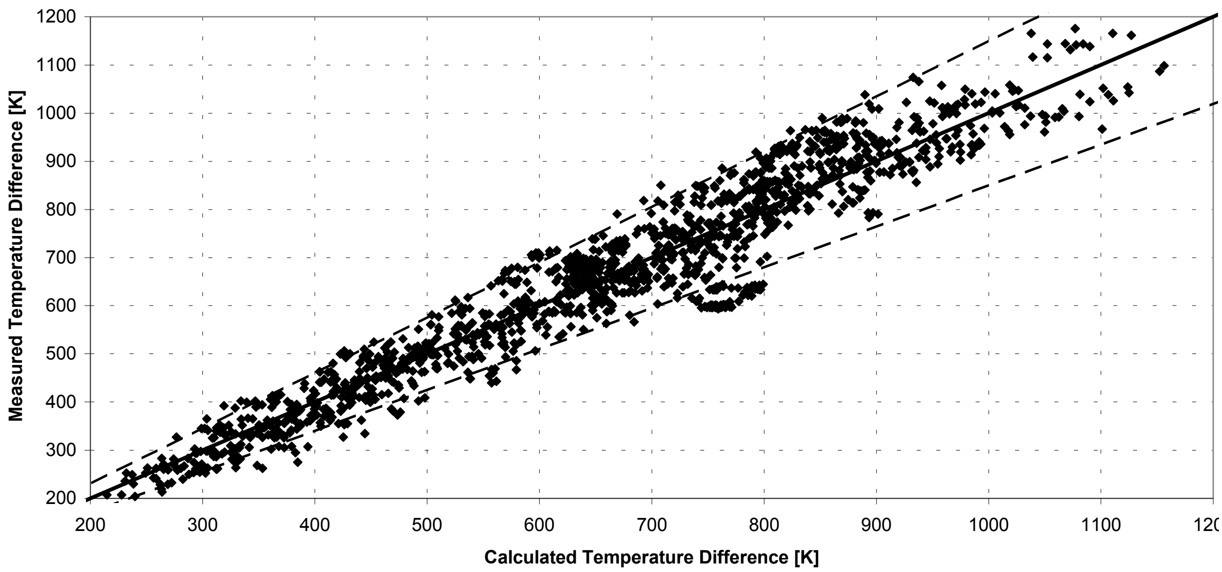


Figure 1. Summarized comparison results of the calculated to measured temperature differences of all He filled SOFIT-1 fuel rods (excluding rod №6) during the entire period of operation

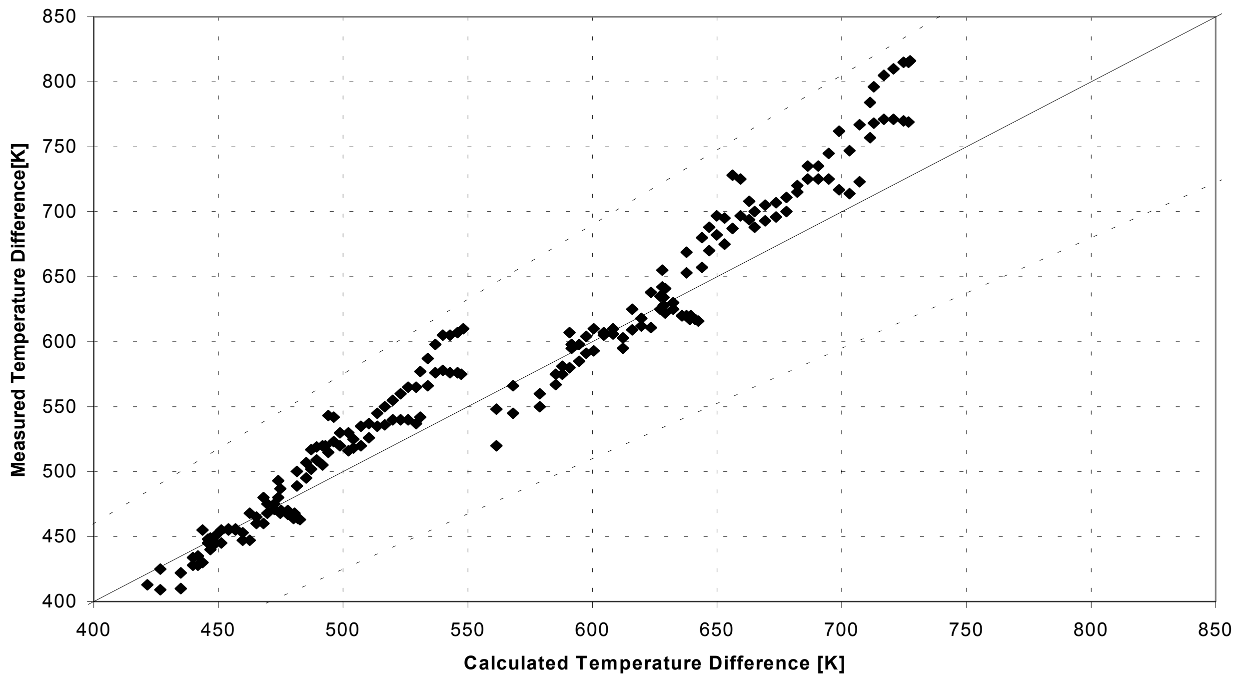


Figure 2. Summarized comparison results of the calculated to measured temperature differences of the four WWER type IFA-503.1 fuel rods during operation at constant power levels of 150 or 200 Wt/cm

in Figures 3-5: the solid one $y=x$, and the dotted $y=0.85x$ and $y=1.15x$.

It is important to note that complete (not "condensed") power histories were used for the code verification against SOFIT-1 data. These data were published in the respective documents (e.g. for SOFIT-11 rod №3 there were 298 calculated points). Condensed power histories were introduced first of all due to the fact that computers existing at the time of SOFIT-1 experimental program were not capable to calculate parameters in all the time intervals.

Figure 1 presents comparison results of the reliable data, only. This data were obtained after evidently wrong data were excluded (after faulty indications of thermocouples and the data on the first power increases). Comparison data on the first power increases are presented in [17] Figure 1.

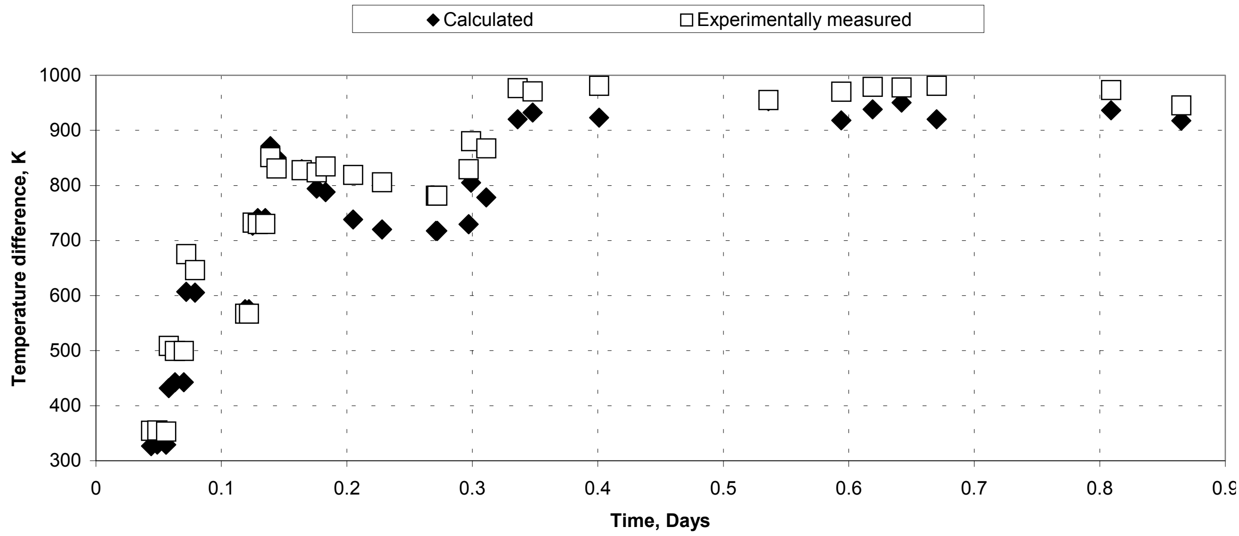
It must be noticed, that during the comparison with the SOFIT-1 data, together with accounting for the measurement errors, the uncertainties [10,16] of the: linear power, 5-7%; fuel rod geometry (including the diameter gap), ±10-20 micron; fuel density and the coolant temperature, have to be also accounted for.

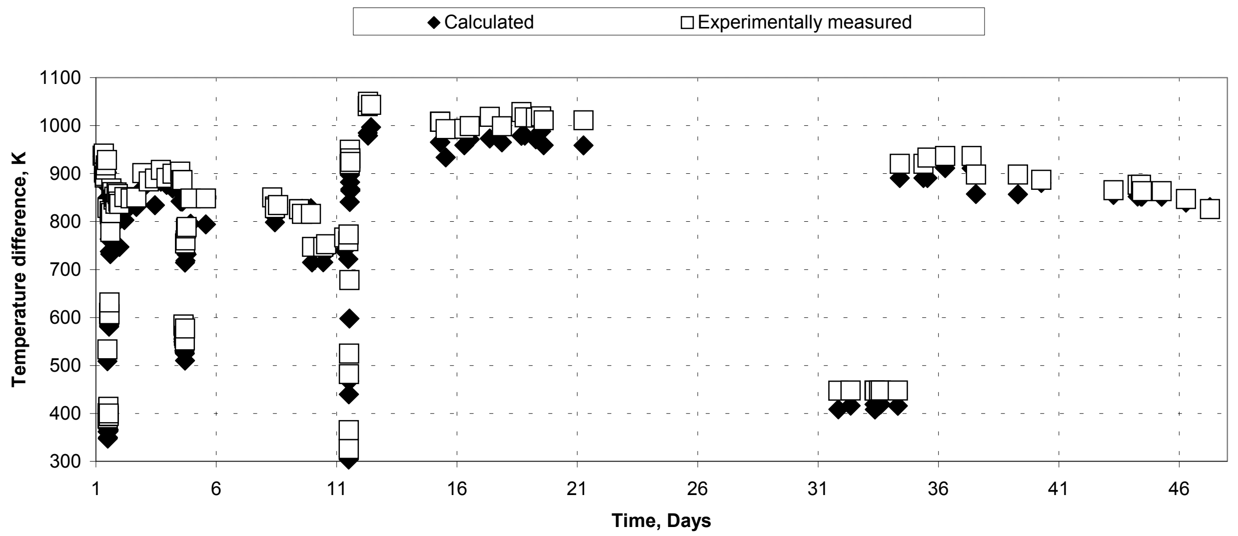
Figure 3 presents experimental and TOPRA-s calculated data on the average in height thermocouple of the SOFIT-1.1 fuel rod №3 [10]. Figure 3 consists of three parts embracing the following time intervals: 0-0.9, 1-48, and 50-195 days.

Experimental fuel assembly with the WWER-type fuel rods has been radiated since 11.12.87 till 18.09.88 (SOFIT-1.2). Six fuel rods were equipped with thermocouples; four of these six were filled with helium. Maximum burnup was ~18, and the average 10.5 MWd/kgU. Figure 4 presents summarized comparison results of the calculated to measured data of all He filled SOFIT-1.2 fuel rods during the entire period of operation.

The TOPRA-s calculated results of the fuel maximum temperature have been compared to the available results of calculations using the codes TOPRA, RET (TR), START-3, ENIGMA. The comparison shows sufficient closeness of the calculated results, in the region of 1-7%. The TOPRA-s testing and validation results prove its applicability for operation calculations of the WWER fuel rod temperature fields. The uncertainty of the obtained TOPRA-s results is subject to the imperfection of the code methodical approach (first of all, when the power history of a particular rod deviates from the "typical" one), as well as to the random factors, such as pellet cracking and relocation, volumetrically power generation and heat removal asymmetry, fuel-to-cladding-to-coolant system geometrical asymmetry, etc.

Besides, the comparison was performed with the results calculated by the code analyzing fuel rods of the power reactors – TRANSURANUS – version for WWER reactors [17]. Figure 5 illustrates the data (calculated with the codes) on the maximum fuel temperature for all 10 zones of one of the ten reviewed fuel rods up to the average burnup in the rod of 45 MWd/kgU. Along the axes are "TRANSURANUS results" – "TOPRA-s results". Altogether, Figure 5 presents the results of 5620 comparisons.





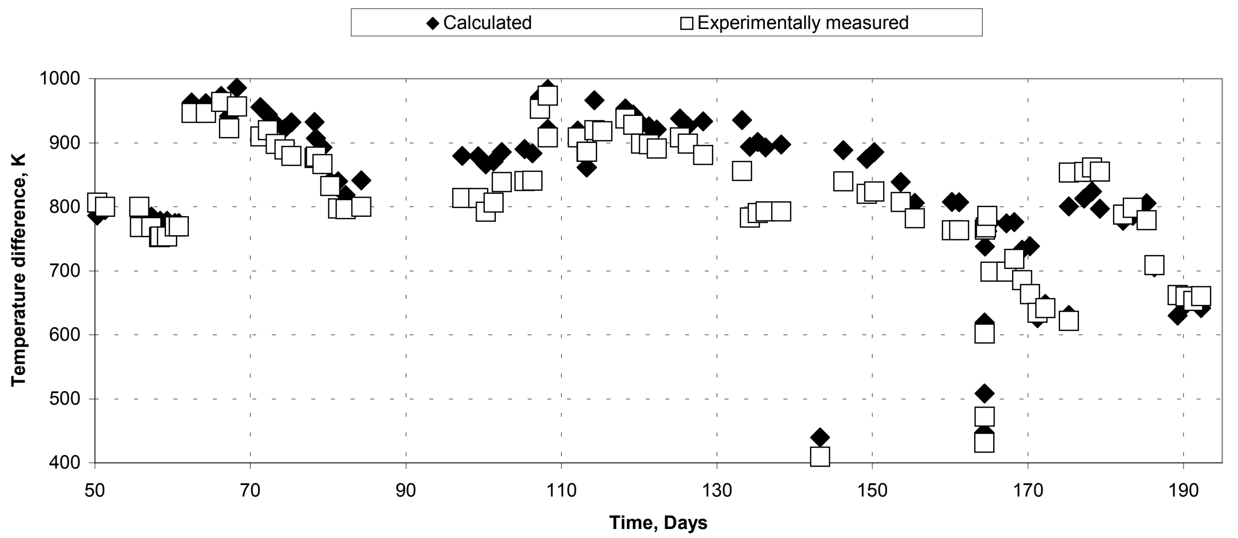


Figure 3. Comparison results of the data regarding SOFIT-1.1 fuel rod 3

As a result the following conclusions were made [17]:

- As a whole, the comparison results show a reasonable agreement of the fuel temperature predictions, taking into account the significant uncertainty of the fuel rod characteristics during irradiation;
- The fuel-to-cladding gap conductance discrepancies in the region of burnup higher than 20-30 MWd/kgU are significant. In this connection it is necessary to notice, that the discrepancies in the gap conductivity in the region of burnup high enough, at values higher than 0.8-1 W/cm^{2}K, insignificantly influence the calculated fuel temperatures. At that, practically all discrepancies can be explained by differences in the:
  - Relocation models;
  - Fuel thermal conductivity correlations versus burnup;
  - Model accounting for the incomplete gas mixing under the cladding at low gap sizes and after fuel-to-cladding contact in TOPRA-s.

Thus, actually all discrepancies can be reasonably explained.

TOPRA-s code has been developed since 1996 [18]. Results obtained in the course of the code verification (1998-1999) indicate the possibility of application of the method and TOPRA-s code for the simplified engineering calculations of thermo-physical parameters of the WWER fuel rods. In November 1999 the application for certification of the code was handed, and in 2001 the code was certified by GAN of RF [11]. The code is intended for calculations of the linear power and burnup typical of the WWER fuel (fuel and gadolinium) rods of the existing fuel cycles, and the cycles that are being justified. Including:

- Linear power not higher than the maximum accepted for the reviewed reactor type;
- Average burnup along the fuel rod cross-section – up to 65 MWd/kgU;
- Average burnup in the fuel rod – up to 60 MWd/kgU;
- Structure, dimensions, and technological margins of the fuel rods are close to the modern or previously used for the WWER fuel rods;
- Coolant temperature – typical of WWER.

Restriction for the code application is, in particular, the fact that the maximum fuel temperature of the analyzed fuel rod does not exceed and did not in the process of operation exceed 1800ºC.

The maximum relative error for predicting fuel rod characteristics in the range of the accepted parameter values is:

- For the difference of the maximum or average fuel temperature and the temperature of the cladding outer surface: ±15%;
- For thermal conductivity of the fuel-cladding gap or contact:
  - Up to the burnup of 15 MWd/kgU: ±20%;
  - At the burnup higher than 15 and less than 40 MWd/kgU – 50/+100%;
  - At the burnup higher than 40 MWd/kgU: ±30%.

TOPRA-s code has been introduced into the KASKAD neutronic code package to generate data on the fuel temperature.

References

[1] A. Scheglov, V. Proselkov, A. Smirnov, et al. Simulation of WWER-440 Fuel Rods Behaviour under High Burnup (Ref. – Unit 3 Kola NPP). Atomic Energy, 81(4), 254-261, 1996.

[2] P. Strijov, F. Pazdera, M. Valach, et al. User's Guide for the Computer Code PIN-Micro. NEA-DATA Bank, November, 1991.

[3] A. Scheglov, V. Proselkov. Some Results of PIN-mod2 Code Verification. Proc. 2-nd International Seminar WWER Reactor Fuel Performance, Modelling and Experimental Support. Sandanski, Bulgaria, 21-25 April, 1997, INRNR – BAS, 1997.

[4] V. Proselkov, A. Scheglov, A. Smirnov, Smirnov V.P. Features of Fuel Performance at High Fuel Burnups. Proc. 11-th AER Symposium VVER Reactor Physics and Reactor Safety. Csopak, Hungary, 24-28 Sept., 2001.

[5] N. Malinin. Applied Theory for Plasticity and Creep. Moscow, Machinistroyeniye, 1975.

[6] A. Tutnov. Methods to Analyze Working Ability of Structural Elements of the Nuclear Reactors. Moscow, Energoizdat, 1987.

[7] A. Scheglov. Influence of the Fuel-Cladding System Deviations from the Model of Continuous Cylinders on the Parameters of the VVER Fuel Element Working Ability. Proc. Int. Sem. VVER Reactor Fuel Performance, Modelling and Experimental Support. St. Konstantine, Varna, Bulgaria, 7-11 Nov., 1994.

[8] A. Scheglov. Influence of the Fuel-Cladding Eccentricity, Cladding Ovality, and Fuel Pellet Chippage onto the Temperature Field in the Fuel Rod. Atomic Energy, 67(3), 204-207, 1989.

[9] A. Scheglov. Influence of the Gap between Fuel Pellets onto the Temperature Field in the Fuel Rod. Atomic Energy, 71(2), 159-161, 1991.

[10] A. Scheglov. TOPRA-s Code to Analyze Thermophysical Characteristics of the WWER Fuel Rod Cross-Sections. Preprint RRC KI, 6172/4, Moscow, 2000.

[11] TOPRA-s Code. Certification passport number ПС No 126 dated 12.04.2001;

[12] V. Sidorenko, et al. Spectral Code TVS-M for Calculation of Characteristics of Cells, Supercells and Fuel Assemblies of VVER-Type Reactors. Proc. 5-th AER Symposium, Dobogoko, Hungary, 15-20 Oct., 1995.

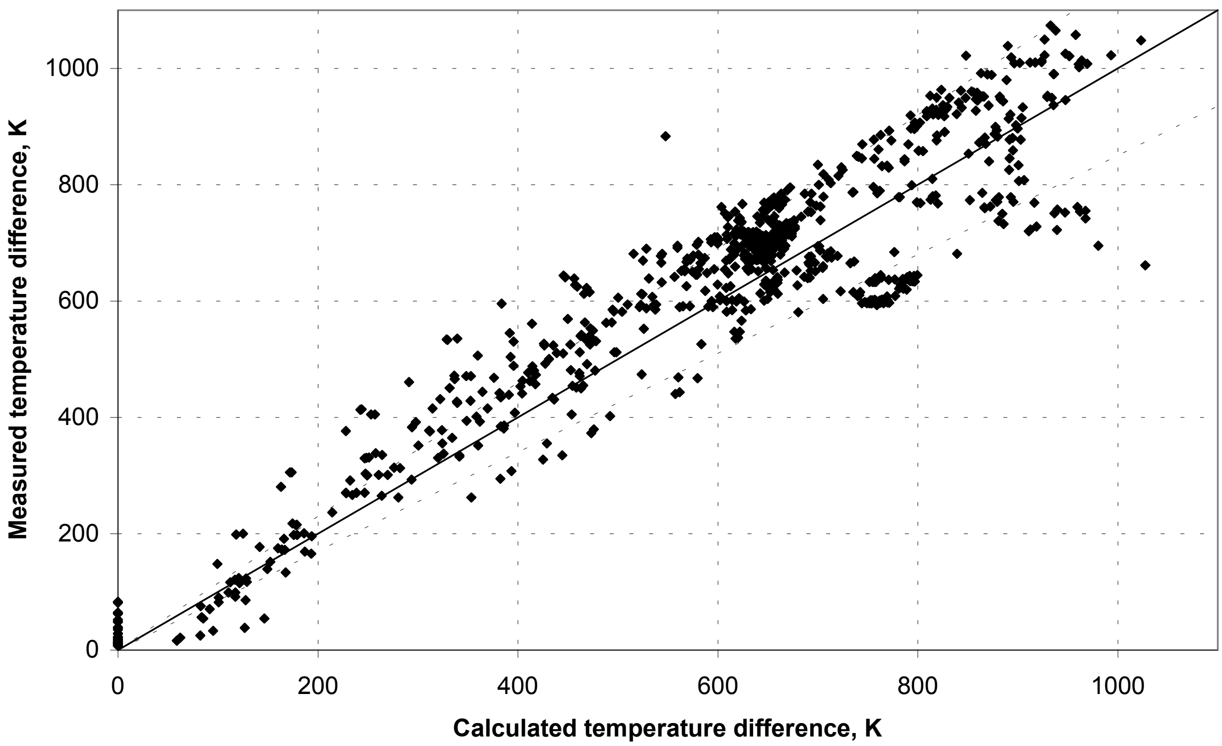


Figure 4. Comparison results of the data regarding SOFIT-1.2 fuel rods

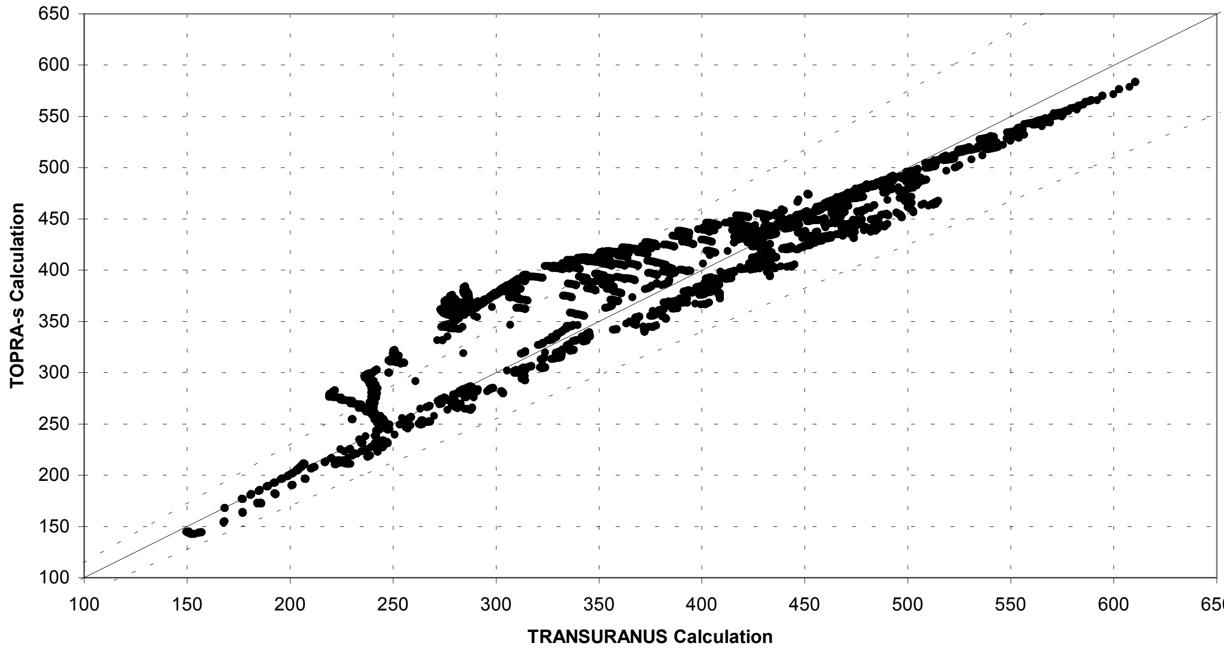


Figure 5. Calculated data on the maximum fuel temperature for all ten zones of one fuel rod

[13] A. Smirnov, et al. Experimental Support of VVER-440 Fuel Reliability and Serviceability at High Burnup. Proc. Int. Seminar VVER Reactor Fuel Performance, Modelling and Experimental Support. St. Konstantine, Varna, Bulgaria, 7-11 Nov., 1994.
[14] A. Smirnov, V. Smirnov, et al. Behaviour of WWER-440 and WWER-1000 Fuel in a Burnup Range of 20-48 MWd/kgU. Proc. 2-nd International Seminar WWER Reactor Fuel Performance, Modelling and Experimental Support. Sandanski, Bulgaria, 21-25 April, 1997, INRNR – BAS, 1997.
[15] M. Solonin, Yu. Bibilashvili, A. Ioltoukhovsky, et al. WWER Fuel Performance and Material Development for Extended Burnup in Russia. Proc. 2-nd International Seminar WWER Reactor Fuel Performance, Modelling and Experimental Support. Sandanski, Bulgaria, 21-25 April, 1997, INRNR – BAS, 1997.
[16] V. Yakovlev, et al. Qualification and Interpretation of MR Test Reactor Irradiation Data on VVER-440 Type Fuel Rods for Fuel Thermal Model Validation. IAEA-TC-659/1.4, 50-56.
[17] A. Scheglov, V. Proselkov, V. Sidorenko, G. Passage, S. Stefanova, Tz. Haralampieva, Tz. Peychinov. Comparative Calculations of the WWER Fuel Rod Thermophysical Characteristics Employing the TOPRA-s and the TRANSURANUS Computer Codes. Proc. 10-th AER Symposium, Moscow, 18-22 Sept., 2000.
[18] A. Scheglov, V. Proselkov. Methods and Approaches to Calculate Dependencies of Thermal-Physical Parameters of WWER-440 Fuel Rods Versus Burnup and Linear Heat Rate. 7-th AER Symposium VVER Reactor Physics and Reactor Safety. Hornitz near Zittau, Germany, 23-26 Sept., 1997.