Presented are the data for computer codes to analyze WWER fuel rods, used in the WWER department of RRC “Kurchatov Institute”. Presented is the description of TOPRA-2 code intended for the engineering analysis of thermophysical and strength parameters of the WWER fuel rod – temperature distributions along the fuel radius, gas pressures under the cladding, stresses in the cladding, etc. for the reactor operation in normal conditions. Presented are some results of the code verification against test problems and the data obtained in the experimental programs. Presented are comparison results of the calculations with TOPRA-2 and TRANSURANUS (V1M1J06) codes. Results obtained in the course of verification demonstrate possibility of application of the methodology and TOPRA-2 code for the engineering analysis of the WWER fuel rods.
INTRODUCTION.

VVERD INR RRC KI is the scientific adviser in the area of the WWER cores. Therefore, it is necessary to have complete set of computer codes for the analysis (or assessment) of the performance of all the elements of the WWER core including fuel rods. VVERD performs additional justification of the working ability of fuel rods and analyzes their physical parameters in the process of operation. A number of computer codes are used to analyze tight WWER fuel rod parameters in the regimes of normal operation: MKK, MRZ, TOPRA-s, TOPRA, and TOPRA-2.

Papers [1, 2] present short descriptions of 2-D codes analyzing some parts of a fuel rod in the r-z and r-φ geometries: MRZ and MKK – are based on the finite element method. Values for the increase of the local heat flux from the fuel cladding due to the peak of the energy generation as a result of the gap in the fuel column with consideration of the heat migration in the axial direction were obtained with the help of MRZ code [2]. These values are used to calculate influence of the margin coefficients onto the fuel rod linear heat rate for all the reactors using WWER fuel.

Papers [1, 2] also present the information about TOPRA-s [3] code intended for the express-analysis of thermophysical parameters of the WWER fuel rod cross-section. Results calculated with the code are used to analyze neutronic characteristics of the WWER fuel cycles, as well as the activity of the fission products released from the fuel inside the WWER fuel cladding. The code has become part of the code packages – design and operational “KASKAD” and BIPR-8A– based code package.

1. TOPRA-2 CODE.

This report presents information on TOPRA-2 code, intended for modeling the parameters of the WWER fuel rods in the regimes of normal operation (under steady-state, transient, and maneuver regimes of the reactor operation). The analyzed parameters are: fuel and cladding temperatures, thermal conductivity of the fuel-cladding gap, pressure and composition of the gas media inside the cladding, release of the fission gas inside the cladding, change of diameters and lengths of parts of fuel, cladding, and the whole of fuel rod, stresses in the fuel and cladding. TOPRA-2 code is intended for performing of operational calculations, justification of working ability of fuel rods in the fuel cycles versus thermophysical and some of strength criteria. It is also intended to analyze fuel rod parameters necessary for the use by other codes as the input data. It is possible that the code becomes a part of the set for “Hortitsa-M” in-core monitoring system.

The code accounts for changes in the conditions of operation (linear heat rate, coolant temperature and pressure, fast neutron flux), for structural and technological parameters of fuel rods, as well as for the main processes that take place during operation and influence the fuel rod performance. These parameters are: heat transfer from cladding to coolant; formation of oxide layers on the cladding surface; radiation densification; thermal expansion; elastic and plastic deformation; fuel swelling, cracking and creeping; creeping, thermal expansion, elastic and plastic deformation, radiation growth of the cladding; fuel-cladding mechanical interaction (FCMI), including pressure of the fuel-cladding contact; fission gas release out of fuel and therefore change of the quantity and composition of the gas media inside the cladding, its thermal conductivity, and the length of temperature drops at the fuel-gas and gas-cladding boundaries; change of the power density along the fuel radius, and thus – burnup distribution along the radius, including the surface rim-effect, change of the fuel thermal conductivity, change of the free volume inside the cladding including due to the changes in the fuel open porosity and geometry of the cladding, fuel, gas plenum.

Methods of the theory of elasticity, plasticity, and creeping are used for strength calculations [4, 5] (the block was developed by Prof. Tkachev and Mr. Zheltukhin). Cladding deformation is calculated according to the streaming theory. Anisotropy of zirconium tubes is taken into consideration. Fuel deformation modeling is performed according to the modified
ageing theory. Bounded (via friction and pressure) non-linear boundary problem of the combined fuel and cladding deformation is solved. Fuel and cladding are analyzed in accordance with the analytical model of the “thick-walled cylinder” as separate elements. Consideration of the fuel and cladding interaction in the radial (parity of contacting radii) and axial (friction, potential slippage and adhesion) directions is performed after beginning of FCMI.

2. VERIFICATION OF TOPPA-2, VERSION 2, CODE.

The code was tested with the tasks of calculating temperatures, stresses, displacements in the fuel and cladding, cladding creeping, fuel swelling, gas pressure, and free volume inside the cladding, effect of axial forces. Good agreement of the numerical calculation results and analytical solutions is demonstrated.

As an example let’s review testing results of the task to calculate stresses and deformations in the cladding due to established creeping. It is assumed that the tube material is noncondensable. Cylinder inner diameter is 7.76 mm, outer diameter – 9.1 mm, internal pressure is 6 MPa. It was assumed that the creeping axial deformation was equal to zero. This is correct in case when axial force only arises due to the pressure onto the bottom. It is assumed that the dependence of the plastic deformation intensity F versus stress S and time t in case of the established creeping is: \( F = S^n \times W(t) \), where \( n = 3 \); it was assumed that: \( W(t) = A + B \times t \), where \( A = 0.25 \times 10^8 \) [1/MPa] \( n \), \( B = 10^{10} \) [1/MPa] \( n \times (1/h) \).

Figure 1 presents distributions of radial and axial deformations, as well as hoop strain along the tube radius, obtained as the result of numerical calculations at the time moment of 5 days. Deformation values due to creeping were found as the difference of deformations: calculated and caused by elastic deformation (occurring at the moment of loading with pressure). The same figure demonstrates results of the analytical solution of the task [4]. It is seen in the figure that numerical and analytical solutions of the task are graphically indistinguishable. Comparison of numerical results of the distributions along the tube radius of the radial, hoop and axial stresses demonstrated good agreement with the similar analytical results.

![Figure 1](image-url)  
Fig. 1. Distributions along the tube radius of the radial (r), hoop (t), and axial (z) deformations obtained as the result of numerical calculations with TOPRA-2 code (pac) and due to analytical solution of the task (ah).

Reviewed is the influence of the number of sections along the fuel and cladding radius onto the calculated values of temperatures, stresses and deformations. It is demonstrated that
segmentation along the cladding radius into 7 zones and fuel radius – into 20 zones results in the accuracy sufficient for the engineering calculations.

Good agreement of results was demonstrated during comparison of the fuel rod thermophysical parameters calculated with TOPRA-2 and TOPRA-s [1-3] codes.

Results calculated with the code were compared with the data of the post-test investigations of WWER fuel rods, performed at FSUE NIIAR and at Loviisa NPP (with the pre-characterized fuel rods). Reviewed were fuel rods from six fuel assemblies of WWER-440 and two fuel assemblies of WWER-1000. Average values of the fuel rods were used in the calculations as the input geometrical and structural parameters. History of the fuel rod operation was produced in the effective days of operation. It was also necessary to compare calculated and measured values of: fuel cladding elongations; diameter average change in the core section; fuel-cladding gap distribution along the height of fuel rods; fission gas release out of fuel inside the cladding; pressure, free volume and composition of the gas media inside the cladding. The values of initial pressure of filling the fuel rod with helium were varied. When the fuel burnup values measured or calculated (with the use of the data of neutronic calculations) differed greatly, Calculation 2 was performed for the same change of the fuel rod linear heat rate versus time so that the calculated burnup were equal to the measured burnup.

Comparison of the measured and calculated elongations of the fuel claddings and fuel columns after 2, 3 and 4 years of operation was performed for the WWER-440 pre-characterized fuel rods. Comparison of the measured and calculated values was also performed for fuel rods of some assemblies when presenting their history in the calendar days of operation.

Figure 2 presents an example of the calculated and measured values of the fission gas release out of fuel inside the cladding of 25 fuel rods of the two operating fuel assemblies (FA) [6] as the function of the average calculated burnup in a fuel rod. Figure 3 presents comparison results of the gas pressure inside the cladding for the same fuel rods; figure 4 presents the same data obtained with Calculation 2 (with the correction of power histories of 5 fuel rods, for which calculated pressure was lower than the measured value; pressure was measured for three of these fuel rods) and with a variation of initial filling pressure – not 6, but 6.3 bars. Figure 5 presents comparison results of the free volume inside the cladding for the same fuel rods (excluding the data for fuel rod # 22 of FA-222, presented below).

Fig. 2. Calculated and measured fission gas release out of fuel rods of fuel assemblies FA-198 and FA-222.
Fig. 3. Calculated and measured values of gas pressure inside the cladding of fuel rods of fuel assemblies FA-198 and FA-222.

Fig. 4. Calculated and measured values of gas pressure inside the cladding of fuel rods of fuel assemblies FA-198 and FA-222. Calculation 2.

Fig. 5. Calculated and measured values of free volume inside the cladding of fuel rods of FA-198 and FA-222.
Analysis of the obtained comparison results was performed; potential errors while measuring the parameters were reviewed. For example, for fuel rod #22 of FA-222 (fig. 2-4) the measured fission gas release is 1.63%, the calculated – 1.303%; the measured values of the diameter decrease (60 µcm), elongation (12.67 mm), volumes under normal conditions for gas (113.4 cm³) and helium (92.94 cm³) are close to the corresponding values of other fuel rods of PK-222. Measured or calculated burnup value for this fuel rod is 48.4 or 49.72 MWd/kgU. But the value of free volume inside the cladding for this fuel rod (8.1 cm³) differs greatly from the volume values of the other fuel rods (10–12.1 cm³ for FA-222, 10.6 – 12.5 cm³ for FA-198). The same high gas pressure inside the cladding (1.4 MPa) was also registered for only one of the reviewed fuel rods – with the maximum burnup. As the gas pressure was calculated with the use of the free volume value, it is possible to assume that a mistake was made while measuring free volume of this fuel rod.

Verification was performed against experimental data with the WWER type of fuel – SOFIT-1, FGR-2, and IFA-503.2 (see [1, 3] for description of experiments).

The first experimental fuel assembly (SOFIT-1 – SOFIT-1.1 program) was irradiated in the MR reactor for 4730 calendar hours, 3780 effective hours. Specific energy generation was: the average for fuel rods of the FA – 8.8 MWd/kgU, the maximum – 15.5 MWd/kgU. Temperature of the fuel center in the cross-section was measured with the help of 7 temperature detectors for 6 fuel rods. Two of these fuel rods were also equipped with the detectors measuring gas pressure inside the cladding. For some fuel rods, not equipped with the detectors, fission gas release inside the cladding was measured after discharge.

Verification was performed against all the data of this experiment. Complete (not “condensed”) power histories of fuel rods were used. As an example, here are the comparison data for thermocouple – central along the height of fuel rod #3 of this experiment. Helium filling pressure for this fuel rod was 0.1 MPa. Fuel-cladding diametrical gap at the thermocouple location was equal to 200 µcm, and the fuel density – 10.66 g/cm³. Fig. 6 presents calculated and measured temperature values for this thermocouple versus time of operation. Figure 6 consists of two sub-figures representing different time intervals of the fuel rod operation.

![Figure 6.1: Comparison results of the measured and calculated temperatures of the central (along the height) thermocouple of fuel rod #3 in SOFIT-1.1 experiment. Time interval: 0.4 – 195.3 days. Here and below: ■ - calculated values, ∆ – measured values, * - linear heat rate at the thermocouple location.](image-url)
Fig. 6.2. Comparison results of the measured and calculated temperatures of the central thermocouple of fuel rod #3 in SOFIT-1.1 experiment. Time interval: 0 – 0.4 days.

Verification was also performed against elongation of the fuel column and cladding of the two fuel rods of SOFIT-1.3 experiment for the first 78 days of operation.

During verification against the data of IFA-503.2 [7] experiment the comparison was made for the temperature of the fuel column center or pressure of the gas media inside the cladding and elongation of the fuel column. As an example, figures 7 and 8 present dependences of the measured and calculated: gas pressure (fig. 7), and temperature of the fuel center (fig. 8) for the fuel rods of IFA-503.2 experiment as the function of burnup.

Fig. 7. Gas pressure comparison results. Fuel rod #20, IFA-503.2 experiment. Here and in fig. 8 “black” indicates measured results, “grey” – calculated results.
The verification against FGR-2 [8] experiment was dedicated to the study of influence of the stepwise power increase with subsequent hold period at each of the reached levels onto behavior of refabricated fuel rods with the burnup of approximately 50 or 60 MWd/kgU. Comparison was made for the temperature of the fuel center, fission gas release, diameter changes of the cladding and of the fuel-cladding gap after experiment.

In the process of verification initial parameters of every experiment were varied within the limits of uncertainty (inaccurate knowledge of the linear heat rate, fuel rod geometrical parameters, readings of detectors, value of the initial helium pressure in fuel rods, etc.). Analysis of the obtained results allowed making a conclusion on the possibility of applying TOPRA-2 code to calculate WWER fuel rod parameters.

3. VERIFICATION OF TOPRA-2 CODE AGAINST CALCULATION RESULTS OF TRANSURANUS CODE.

Let’s review some comparative results of the calculations made with TOPRA-2 and TRANSURANUS (V1M1J06) codes. TRANSURANUS is a computer code for the thermal and mechanical analysis of fuel rods in nuclear reactors [9–12]. It was developed at the Institute for Transuranium Elements (ITU) and extended to calculate WWER fuel rods in the frames of several EC projects [11, 12]. The TRANSURANUS code consists of a clearly defined mechanical-mathematical framework into which physical models can easily be incorporated. Besides its flexibility for fuel rod design, the TRANSURANUS code can deal with a wide range of different situations, as given in experiments, under normal, off-normal and accident conditions (LOCA, RIA). The time scale of the problems to be treated may range from milliseconds to years. During its development of the code great effort was spent on obtaining an extremely flexible computation tool, which is easy to handle and exhibits very fast running times. The code has a comprehensive material data bank for oxide, mixed oxide, carbide and nitride fuels, zircaloy Zr1Nb (N1) and steel claddings and several different coolants. It can be employed in two different versions: as a deterministic and as a statistical code.
In 1994–2006 the TRANSURANUS code has been extended and verified for WWER type fuel rods [11, 12]. It is important to note that comparative analysis of the calculated results obtained with TOPRA-s / TOPRA-2 and TRANSURANUS codes has been performed [13, 14].

Fuel rod #7 of PK-222 [6] was selected for comparative analyses. Power history of this fuel rod was extended, thus increasing the operation time by 9%. Burnup of this fuel rod was correspondingly increased – of up to ≈59 MWd/kgU. The option of approximately average (in the tolerance range [6]) initial parameters was reviewed. Comparison was made for both the fuel rod integral parameters (elongation, gas pressure inside the cladding, and FGR) and for the parameters of elevation cross-sections: temperature distribution along the radius, thermal conductivity of the gap, change of the radius of the cladding external and internal surfaces, strains in the cladding.

Comparison demonstrated reasonable agreement of the calculated results. As an example let’s review the comparison results of the gas pressure inside the cladding of the fuel rod. Figure 9 (consisting of two parts) presents the calculated results of the gas pressure obtained with the help of both codes.

![Figure 9.1](image1.png)

Figure 9.1. Calculated values of the gas pressure inside the cladding of the fuel rod.

![Figure 9.2](image2.png)

Figure 9.2. Calculated values of the gas pressure inside the cladding of the fuel rod during the first year of operation.
At the beginning of the fuel rod operation values of the gas pressure, calculated with TRANSURANUS code are close but slightly exceed the values obtained with TOPRA-2 code (fig. 9.2). Difference is very similar during the major part of the fuel performance. If the operation time exceeds 1350 days (fuel bumup in the fuel rod is higher than 50 MWt×day/kgU), gas pressure values calculated with TRANSURANUS code become lower than the values calculated with TOPRA-2 code. Difference becomes larger along with the growth of bumup and reaches 1 MPa at the end of the fuel performance. At the end of the fuel performance under normal conditions gas pressure – if calculated with TOPRA-2 code becomes equal to 1.825 MPa, and 1.416 MPa – if calculated with TRANSURANUS code.

Obtained data were analyzed. Difference of the calculated gas pressure at the end of operation is explained by the higher fission gas release out of fuel calculated with TOPRA-2 code (5.87%) that with TRANSURANUS code (2.99%). The difference can also be explained by a greater relative change of the free volume inside the cladding, when calculated with TOPRA-2 code, than in case the calculations are performed with TRANSURANUS code. When calculated with TRANSURANUS code, values of free volume inside the cladding are: 14.4/13.4/10.0/10.2 cm³ for the cases of: initial cold state/ initial period of operation/ at the end of operation/ after operation at 20°C. When calculated with TOPRA-2 code the corresponding volume values are: 16.0/15.1/10.2/10.6 cm³. It is important to note good agreement of the changes of the free volume values calculated with both codes during the power increase at the beginning of the fuel rod operation (difference of the second and first values) and during the power decrease at the end of operation (difference of the third and forth values). During the time of operation the free volume value calculated with TRANSURANUS code changed by 3.46 cm³ (or by 26% of the value at the beginning of operation). At the same time calculations with TOPRA-2 code indicated that this change was 4.96 cm³ (or by 33%). The difference in the change of the free volume value is mainly due to lower reduction of the cladding internal surface diameter as a result of creeping obtained with TRANSURANUS code.

Difference of the fuel temperature can be another explanation of the difference in the calculated values of gas pressure. As an example, let’s review the data in the axial zone #6 of the fuel rod (when the fuel column is divided into 10 zones of equal height). Figure 10 presents values of the fuel maximum temperature calculated with both codes for this zone (figure 10 consists of two parts).

Figure 10.1. Calculated values of the fuel maximum temperature for the zone #6.
Figure 10.2. Calculated values of the fuel maximum temperature for the elevation zone #6 during the first year of operation.

At the beginning of the fuel rod operation fuel temperature values calculated with both codes are approximately equal. At the very initial moment the value calculated with TOPRA-2 code is by 5 K higher than the value calculated with TRANSURANUS code. During further operation the values, obtained with TRANSURANUS code are a little (by 5-20 K) higher than the ones, calculated with TOPRA-2. Later, along with the burnup growth calculated temperature values come closer. At the end of operation temperature values calculated with TRANSURANUS code get lower (up to 87 K) than the values calculated with TOPRA-2 code. This can be explained by lower (calculated with TRANSURANUS code) value of thermal resistance of the fuel-cladding contact, differences in fission gas release, etc. Higher values of the: fission gas release, relative reduction of the volume inside the cladding, fuel temperatures obtained with TOPRA-2 code in comparison with TRANSURANUS code lead to the noted difference of the gas pressure behavior inside the cladding.

As an example of other comparisons figure 11 presents the data on tangential (circumferential) stresses in the fuel rod cladding for zone #6. These stress values are average for the whole thickness of the cladding (in TOPRA-2 code the cladding is divided into 7 zones along the radius, in TRANSURANUS code – into 5 zones).

Figure 11. Tangential stresses along the cladding thickness in zone #6 versus time.

The data presented in fig. 11 are prior to the moment of the beginning of FCMI as calculated with TOPRA-2 code. After beginning of FCMI stresses in the cladding increase quickly. TRANSURANUS code predicts a later gap closure. After beginning of FCMI stresses calculated with TRANSURANUS code also grow quickly – like the ones calculated with
TOPRA-2 code. The comparison results for stresses indicated good agreement of parameters, calculated with the two codes.

CONCLUSIONS.
Presented is the information on TOPRA-2 code and its verification. Some results of the code verification are also presented. The conclusion is made about the possibility of applying the code to calculate parameters of the WWER fuel rods. Some results and comparison of results of calculating parameters of the WWER-440 fuel rods with TOPRA-2 and TRANSURANUS (V1M1J06) are presented. The conclusion is made about good agreement between calculated results made with both codes.

REFERENCES
6. Scheglov A.S., Proselkov V.N., Bibilashvily Yu.K., Medvedev A.V., Novikov V.V. Corrected data base on the initial parameters, and irradiation history of the fuel rods of the fuel assemblies #198 and 222 irradiated at Kola-3 for 4 and 5 years, respectively, and some data on the post-irradiation examination of the fuel rods of the above fuel assemblies. OECD/NEA/IAEA IFPE international data base. NEA–1532/02 March 1999. 4 p.