# Comparative calculations and operation-to-PIE data juxtaposition of the Zaporozhye NPP, WWER-1000 FA-E0325 fuel rods after 4 years of operation up to ≈49 MWd/kgU burnup

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## 1. Introduciton

Operational and PIE data for the Zaporozhye NPP, FA-E0325, WWER-1000 fuel rods were provided in the OECD NEA IFPE Database and were used to perform comparative calculations among several fuel performance codes. The fuel rods had been irradiated for 4 years of operation up to ≈49 MWd/kgU burnup. The fuel rod operation histories are developed for the PINw99, TRANSURANUS (V1M1J03) and TOPRA-2 codes. The initial state fuel rod parameters are analysed and calculations are carried out. The PIE data enable the comparison of experimental measurement with code-calculated values for cladding elongation (49 rods), FGR and gas pressure (35 rods). Cladding diameter creep-down and gap closure results are juxtaposed as well. The capability of the applied codes correctly to predict the WWER fuel rod performance is shown. The authors gratefully acknowledge the OECD-NEA and IAEA for the assistance to obtain the data.

WWER-1000 fuel rod data (further called “data”) have been delivered by the IFPE OECD-NEA Data Bank [1, 2]. The data include initial geometrical and design parameters of the fuel rods, as well as description of the operation regime, NPP unit loading history and PIE results (further called “measured results”) at normal conditions. The data are sufficient for modelling all 312 fuel rod and for comparison of calculations with experimental results for a limited number of fuel rods.

## 2. Fuel rod power histories and design parameters

The E0325 fuel assembly (FA) has been operated at the Zaporozhye NPP, Unit 1, during four fuel cycles (No.4 – No.7), achieving 48.94 MWd/kgU burnup in average. The operation duration was 1142 full power days. All fuel rods are 4.4% enriched by $^{235}U$. The unit power history contains many shutdowns and periods of decreased power level operation. More than the half of the last fuel cycle duration the reactor operated at 60% nominal power due to the custom electric power requirements.

Each of the power histories of 312 fuel rods consists of 403 time points with power given for 10 axial zones along the rod height, the linear heat rate (LHR), the fast neutron fluence (FNF), the cladding surface temperature and burnup.

Of the 312 fuel rods, fifty-five (55), for which PIE results are available, have been selected for the present calculations. Comparisons are made for burnup, FGR, internal gas pressure, gas free volume under the cladding, average cladding diameter decrease in the fuel rod active part, cladding elongation and diametrical fuel-to-cladding gap.

The power histories for the calculations have been developed to comply with the input format requirements of the applied computer codes and can be characterised as follows:

- As the data first time point (time = 0. d) for each fuel rod IFPE power history is given in hot state, an additional time point of initial cold (20°C) or cold zero power state, 0.1 MPa coolant pressure, and zero LHR was introduced. Then the second time point (time = 0.1 d) was assumed to correspond to the data first time point. It was assumed

during the calculations by all codes applied, that the parameters (LHR, cladding temperatures, etc.) change linearly between the first and the second time points, as well as between any other two points;
- Similarly, a last time point was introduced with parameters equal to those of the first time point;
- The data contain two time points representing the reactor reloadings: one before and one after the reloading, with equal time values. For the development of the power histories, in order to avoid steep parameter changes, it was assumed, that the transition time span between the points is 4 hours. A linear approximation of the parameter changes was used;
- The fast neutron flux values were obtained from the data FNF time values along each axial segment. It was assumed, that for each time span any segment fast neutron flux is constant and equals the time span FNF difference divided by the time span duration.

In order to account for the difference between the data FNF with E > 0.1 MeV and the required by the codes FNF with E > 1 MeV (used in the cladding creep and irradiation growth correlations), a dependence has been applied, developed on the base of the HELIOS code [3] with a correction by the RRC "KI" [4].

A correction has been introduced into the LHR values, based on the following. The data contain several burnup values for each rod. For instance, considering a randomly chosen fuel rod No.11, the measured burnup is 45.98 MWd/kgU. In the header of the 011his.txt data file, the given burnup is 46.36 MWd/kgU. The averaged axial segment rod burnup at the end of operation is 47.26, or 47.28 MWd/kgU, accounting for the given axial segment inequality. The value 50.95 MWd/kgU can be obtained by summing along all time spans the products of the rod average LHR and the time span, divided by the rod uranium mass (1.465×0.8795, where 1.465 is the rod fuel mass in kg and 0.8795 is the uranium content in the fuel – a minimum estimation). Thus, the rod average LHR was obtained as the average along the time intervals of the averaged along the axial segments LHR, multiplied by the given 3.536 m fuel stack length.

For the FA average burnup a value of 48.9 (48.94) MWd/kgU is given in the data. The average of all 312 rods, given in the header of the power history files, is 45.959 MWd/kgU. The calculated average of all 312 rods equals 50.526 MWd/kgU (50.95 MWd/kgU for rod 11). The results of the exact neutron-physical calculations carried out at the RRC "KI" give a FA average burnup of 49.5–50 MWd/kgU, on the base of which all LHR data have been reduced by 2% for the presented calculations.

The data contain the relevant fuel rod design and geometry parameter values with the maximum and minimum possible deviations. During the preparation of the calculation inputs these values have been corrected according to the statistical analysis results of the parameter measurement study [5, 6] of WWER-1000 fuel rods, produced by the Novosibirsk Chemical Concentrates Plant in 1989 – 1990, i.e. a couple of years later than the FA given in the data. The following initial parameters have been chosen:

- Fuel stack: length 3.55* m, fuel mass 1465 g;
- Cladding: outer diameter 9.13 mm, inner diameter 7.755 mm;
- Fuel pellets: outer diameter 7.55 mm, inner diameter 2.3 mm, enrichment 4.4%, initial density 10.585 g/cm$^3$, grain size 6 micron, maximum densification 0.8%;
  Note the misprint of the fuel pellet outer diameter in the data;
- Initial filling gas (helium) pressure 2.0 MPa, fuel rod upper plenum volume 11.4 cm$^3$.

* – see below in the discussion of the burnup differences.

## 3. Computer codes used

The comparative calculations have been performed by three computer codes for WWER fuel rod performance analysis: PINw99, TOPRA-2 and TRANSURANUS (V1M1J03).

The code PIN-micro (1990 version) [7] is a simple code for WWER and PWR fuel rod thermomechanical predictions for operational purposes. It has been transferred to the OECD NEA Data Bank in 1991 and has been successfully used since then at NRI (Czech Republic), RRC «Kurchatov Institute» and INRNE (Bulgaria). A number of changes and additions have been introduced into the code during the years and these changes resulted in several versions of the code. The newer improved version of the PIN-micro code developed was named PINw99 [8]. Calculations of fuel rods from SOFIT and KOLA-3 experiments were performed by PINw99 using the data included in the renewed version of NEA Data Bank database [12]. The calculated results were compared with experimental values available, their good agreement justified the applicability of PIN99w code for WWER fuel analyses.

The TRANSURANUS (V1M1J03) code [9, 10] is a code for thermal and mechanical analysis of fuel rods of various designs (LWR, WWER, FBR) at various operational conditions (steady-state, transient, accident). The most significant features of the code are its clear and user friendly code structure, the clearly defined mathematical framework, the high flexibility regarding fuel rod design and time scales of irradiation conditions, the possibility to model different reactor operation regimes and the comprehensive set of material properties for oxide, mixed oxide, carbide and nitride fuels, zircaloy and steel claddings and different coolants. For the present study, the standard LWR version with WWER specific models and options has been applied.

The TOPRA-2 code [11] is developed for modeling of thermophysical and mechanical (strength) characteristics of uranium dioxide and uranium-gadolinium fuel rods of the power and research reactors. It is intended for justification of their working ability in quasi-steady state regimes of operation. Methods of the elasticity, plasticity and creeping theories 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. The 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 analysed as separate elements. Coupling is considered via inner gas pressure, fuel-to-cladding contact pressure and the friction force through friction and pressure.

## 4. Calculated results

In Figures 1–5 the calculated results for rod No.148, one of those having achieved the highest burnup, as a function of time, are shown:

- average and maximum LHR (Fig. 1);
- fuel central temperature, 5$^{th}$ axial segment (Fig. 2);
- fission gas released from fuel (Fig. 3);
- inner gas pressure (Fig. 4);
- cladding elongation (Fig. 5).

In Figures 3–5 the measured values (one point on each graph) are given too.

Rod No.148 is the one with the highest measured discharge burnup of 51.33 MWd/kgU. The corresponding burnup calculated by means of the TOPRA-2 is 54.37 MWd/kgU. The calculated for rod No.302 is greater, 54.95 MWd/kgU, whereas the measured value for this rod is 49.39 MWd/kgU. This must be kept in mind when considering the comparative results shown in Figures 6–10.

Figures 6–10 present the juxtaposed measured and calculated (for normal conditions) by the three codes values as a function of discharge burnup:

- fission gas released from fuel (Fig. 6);
- inner gas pressure (Fig. 7);
- inner gas free volume (Fig. 8);
- cladding diameter change of the active axial part (Fig. 9);
- cladding elongation (Fig. 10).

The presented results concern the fuel rods, for which measured values are available. Because of the difference between the measured and the calculated burnups (discussed above) and for the seek of a better visualness, the burnup values on Figures 6–10 have been taken from the TOPRA-2 calculations.

## 5. Discussion of the results

**Burnup.** For all fuel rods the measured burnups are less than the calculated. For the results obtained by TOPRA-2, the difference varies between 4.5–10%, in average ≈7–8%. A closer difference of 5–7% can be observed if juxtaposing the measured, 46.5–46.9 MWd/kgU, and the calculated, 48.94–50 MWd/kgU, FA average burnup. It should be noted, that a similar systematic lower measured burnups have been observed for the Kola-3 WWER-440 FA198 and FA222 [12].

The burnups calculated by the three codes also differ. The TRANSURANUS values are 3.7–3.8% lower and the PINw99 are 1.3–1.4% lower than the calculated by TOPRA-2. At that, the calculated by TOPRA-2 are based on the minimum possible uranium content in fuel (0.7895), which leads to burnup increase. Small deviations may result from the differences in the fuel stack length used in the different codes:

- the value of 3.536 m was used in the neutron physical calculations, as given in the data;
- the value of 3.54–3.55 should have been used in the neutron physical calculations;
- the value of 3.53 was used in the initial fuel rod cold state.

**Fission gases released from fuel.** The relative low values of the measured FGR (< 0.7%) for most of the FA0325 fuel rods can be explained by the moderate to low levels of LHR, especially at the end of reactor operation. It should be kept in mind when analysing the results that the possible underestimated measured burnups may lead to lower measured FGR (by 5–8%). The FGR calculated by PINw99 and TOPRA-2 overpredict the measured values. The values predicted by TRANSURANUS are in good compliance with the measured ones up to burnups of ≈51 MWd/kgU and lower at higher burnup. A specific feature of the PIN code series, including PINw99, is the BOL ~ 0.5% FGR, which reflects an older conservative concept incorporated in the codes. The recent version PIN2K [16] applied by the INRNE authors in the FUMEX-II exercise [17], gives much better results.

The calculated by all codes FGR values of two rods having achieved lowest burnups are significantly higher than the measured ones. The rather significant uncertainty of the gas composition measured results, which obviously increases at low FGR, must also be accounted for. The calculated by the START code FGR (see data and [13]) are overpredicted as well. Indeed, the difference is less – 0.3 and 0.2% for rod No.34, and 0.24 and 0.19% for rod No.163.

**Inner gas pressure.** As a whole, accounting for the rather significant uncertainties [13] of the gas pressure determination, the results obtained by the TRANSURANUS and TOPRA-2 codes are close to the measured ones, constantly slightly underpredicting. The difference by TOPRA-2 is ≈ 0.05–0.15 MPa, or ≈ 5%, and ≈ 10% by TRANSURANUS. The calculated by PINw99 pressure is higher and deviates by ~ 0.2 MPa the measured values. The differences can be explained by the measurement uncertainties and by the possible initial fill gas pressure deviation from the taken in the calculations 2 MPa. Also, the underpredicted by TOPRA-2 and TRANSURANUS cladding diameter change values may lead to the same trend.

Analyzing the gas pressure calculated results it should be kept in mind, that the absolute values of FGR are low compared to the initial fill gas mass. This leads to a 5% (for most fuel rods < 0.5%) contribution of FGR to the inner gas pressure. The measured results of the helium quantity for most rods at EOL is > 94.8%. This can explain the fact that the differing calculated and measured FGR values do not lead to significant differences in the gas pressure.

**Inner gas free volume.** The TOPRA-2 calculated results show very good agreement with the measured values (taking into account the possible deviations of the real fuel rod geometry from the average parameter values). The calculated by TRANSURANUS gas free volume values systematically underpredict (by 1.5–3 $cm^3$, or 5–10%) the measured ones. Though, having in mind the significant measurement uncertainties, the deviation is relatively small. It should be noted, that the START-3 predictions (see data and [13]) of the gas free volume reveal a similar underprediciton. The calculated PINw99 results are significantly lower (by 5–7 $cm^3$) than the measured values, which leads to significant deviations of the gas pressure too.

**Fuel-to-cladding gap at the fuel active height.** It must be pointed out, that the validation of the codes upon this parameter requires several model assumptions, resulting from the onset of the cladding-to-fuel contact, "soft" and "hard", proven experimentally and in the calculations. In the so-called "soft" contact the cladding touches the fuel pellets, but the fuel cracks remain intact, which partially softens the fuel-cladding mechanical interaction. The "hard" contact occurs after the fuel cracks have been closed backwards (the relocated fuel fragments have moved back) and the fuel may be considered as a continuum (intact). In this case, the fuel-to-cladding gap model is based on an extension of the relocation model, applied also in many other codes. The gap value after the hard contact depends only on the operation conditions during the last 20–40 days of operation – contact pressure and cladding creep rate. All three codes predict full gap closure at working conditions and open gap of 5 microns and more, at normal conditions, which correlates satisfactorily with the measured values (4–44 micron). The calculated by TRANSURANUS values are 3–12 micron, by TOPRA-2 are 6–9 micron and by PINw99 give 10.8 micron, which reflects the fuel and cladding roughnesses.

**Cladding diametrical change.** Having in view the initial cladding diameter uncertainty of +80 micron/–50 micron, the cladding diameter change has been sought experimentally as the difference between the diameters at rod upper plenum and the fuel rod active part height with maximum burnup. Accounting for the measurement uncertainty of ±10 micron, the juxtaposition of this parameter may be observed rather as a qualitative one. As a whole, the measured and the calculated values are in good agreement. The PINw99 and TRANSURANUS results give a smaller diametrical change (–52 ÷ –57 micron). The TOPRA-2 calculated results (–60 ÷ -70 micron) give a more reasonable compliance with the measurement (–50 micron for one fuel rod and between –60 and –80 for the rest of the rods).

**Cladding elongation.** The initial cladding length is known with an uncertainty of ±2 mm and the measurement error is ±0.3 mm [14]. Thus, ± 2.3 mm can be considered as the actual measurement uncertainty. Taking it into account, the TRANSURANUS and TOPRA-2 calculated results could be considered as being in good compliance with the measured ones. Most of the calculated by TRANSURANUS results are by 2–6 mm higher than the measured ones and have a trend to increase with burnup increase. The TOPRA-2 results are in better agreement and have a different trend of decreasing with burnup after the onset of a hard contact. This is a consequence of the applied "axial force" model, which implies prevailing cladding shortening, due to its diameter decrease after PCMI, over the elongation due to irradiation growth and swelling extension by the fuel at hard contact [15]. This means, that before PCMI, the fuel rod elongation is governed by its thermal state, by the anisotropic irradiation growth and by the cladding diameter creep down. After PCMI, due to the reverse diameter change (increase), the fuel rod begins to shorten, and due to the axial force appearance – to elongate.

The calculated by PINw99 elongation values are lower than the measured ones by 10–15 mm, which can be explained by the lack of accounting for the contribution of the cladding diameter creep down and the appearing at PCMI axial force.

## 6. Conclusions

As a whole, from the comparison of the calculated and measured results as discussed above, it can be concluded, that the codes PINw99, TRANSURANUS and TOPRA-2 used in this study, are capable of adequate predicting the thermophysical and the mechanical performance of the WWER-1000 fuel rods. The PINw99 code predicts conservative BOL FGR values and conservative gas pressure values in the region of burnups higher than 30 MWd/kgU, which can be explained by the underprediction of the cladding gas inner volume and cladding elongation. The improved version PIN2K (not applied in the present study) predicts much better FGR and gas pressure [17], though, it is still under development in the high burnup FGR modelling part. In the TRANSURANUS code, there are also areas, where refinements are clearly indicated. They are subject of the ongoing research projects and further improvements are already proposed [18].

## 7. References


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**Figures**

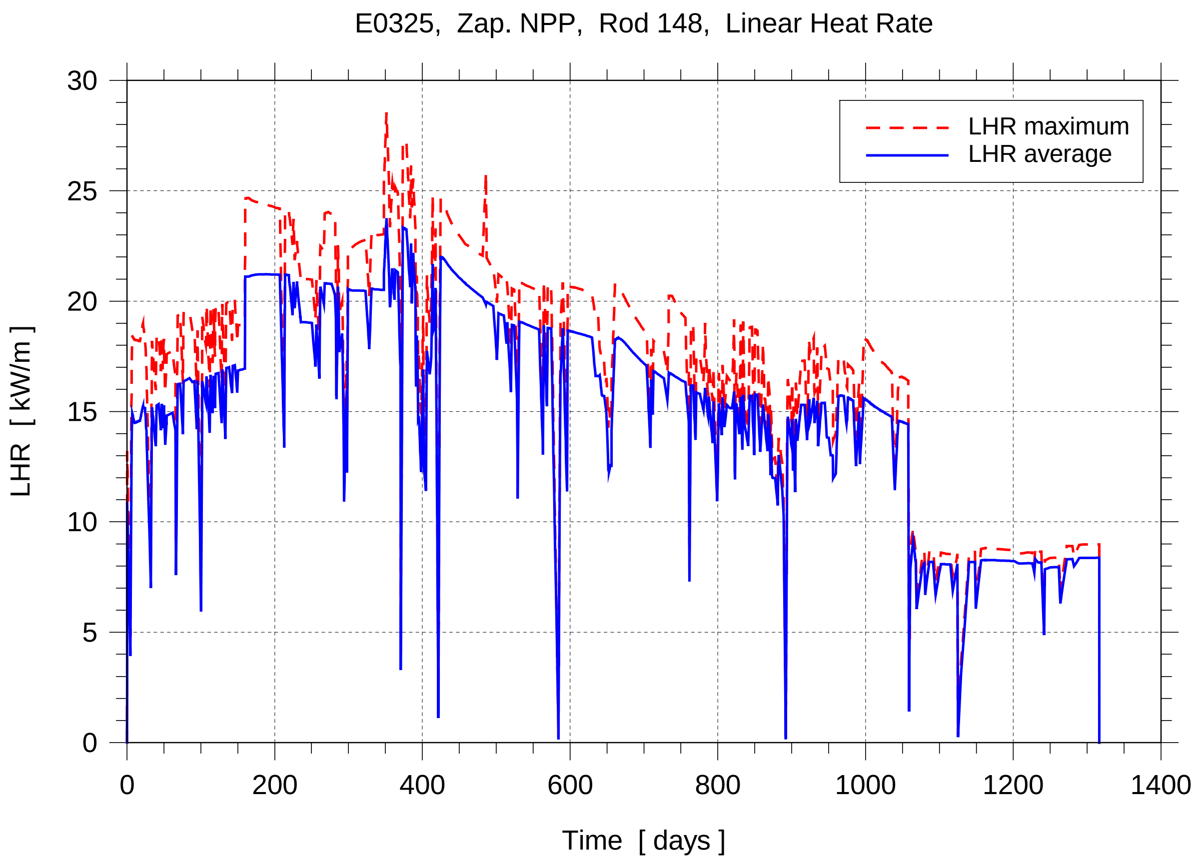


**Fig. 1.** Rod No.148 average and maximum LHR

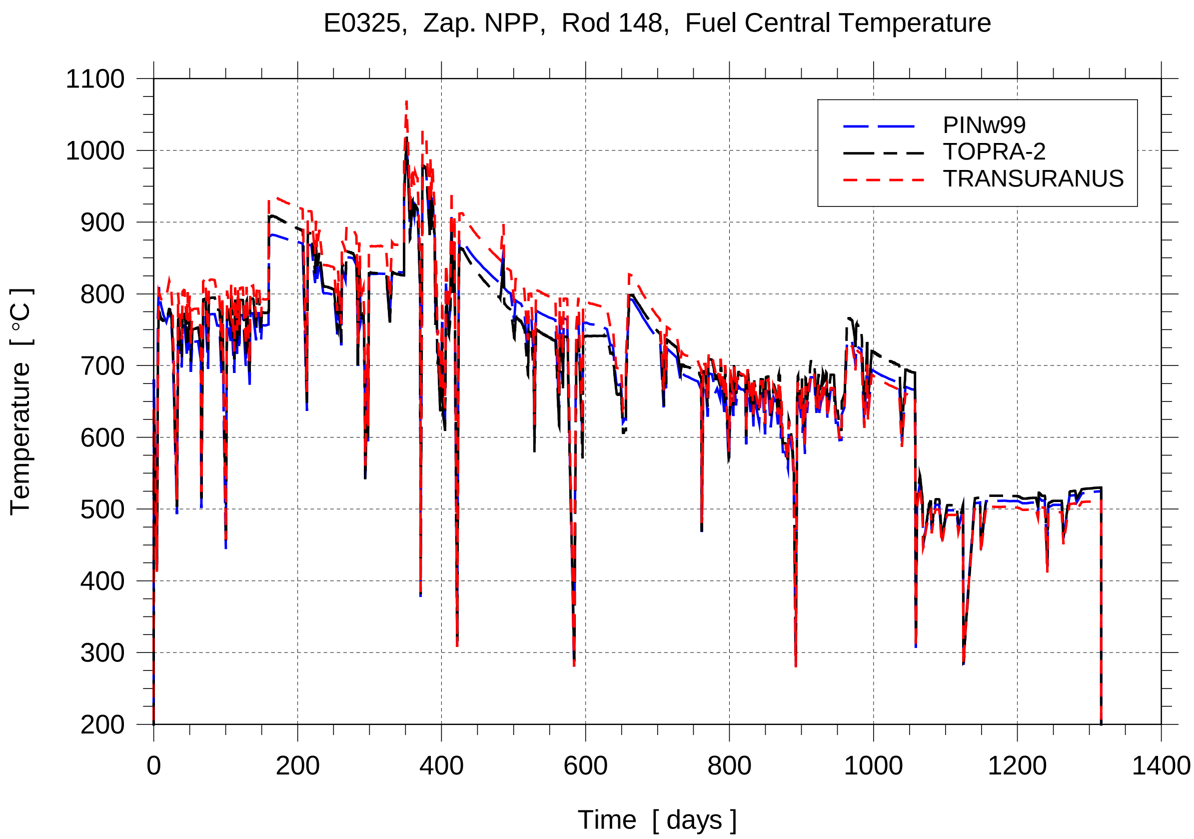


**Fig. 2.** Rod No.148 fuel central temperature, 5th axial segment

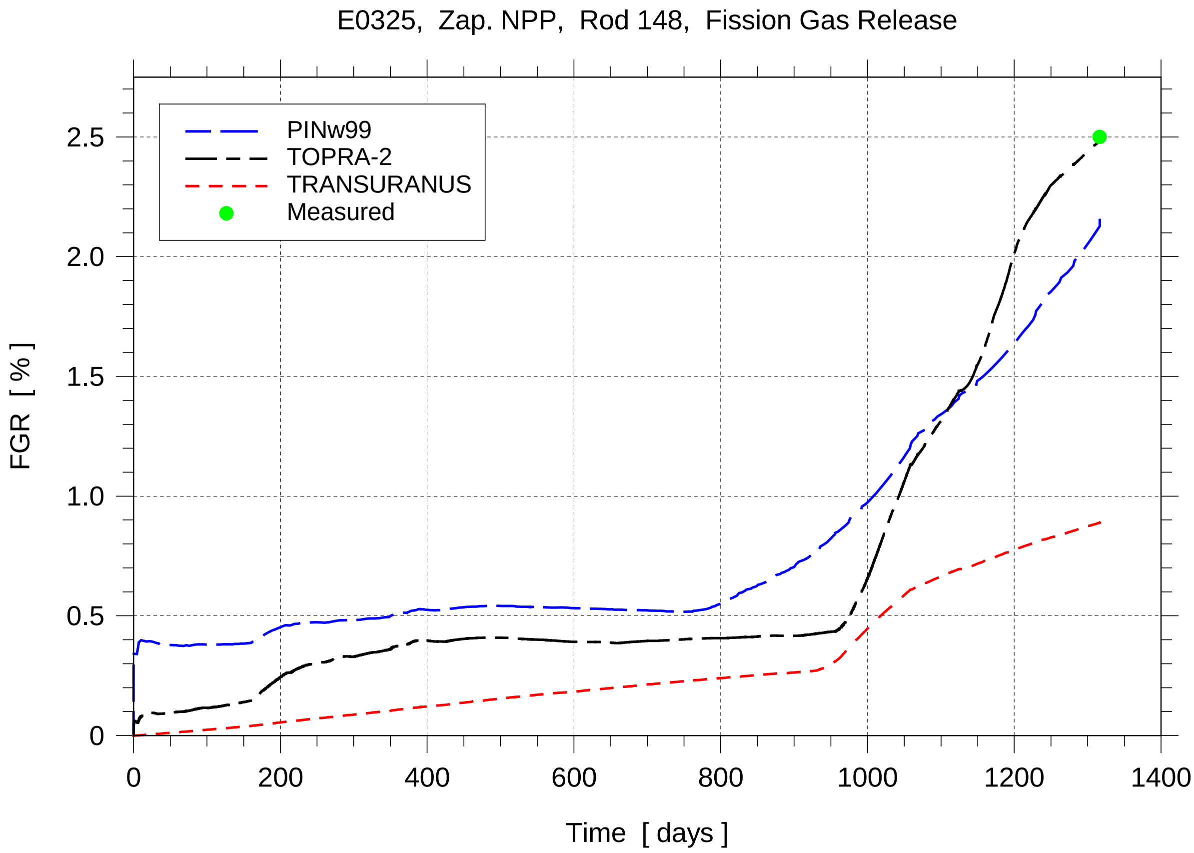


**Fig. 3.** Rod No.148 fission gas released from fuel

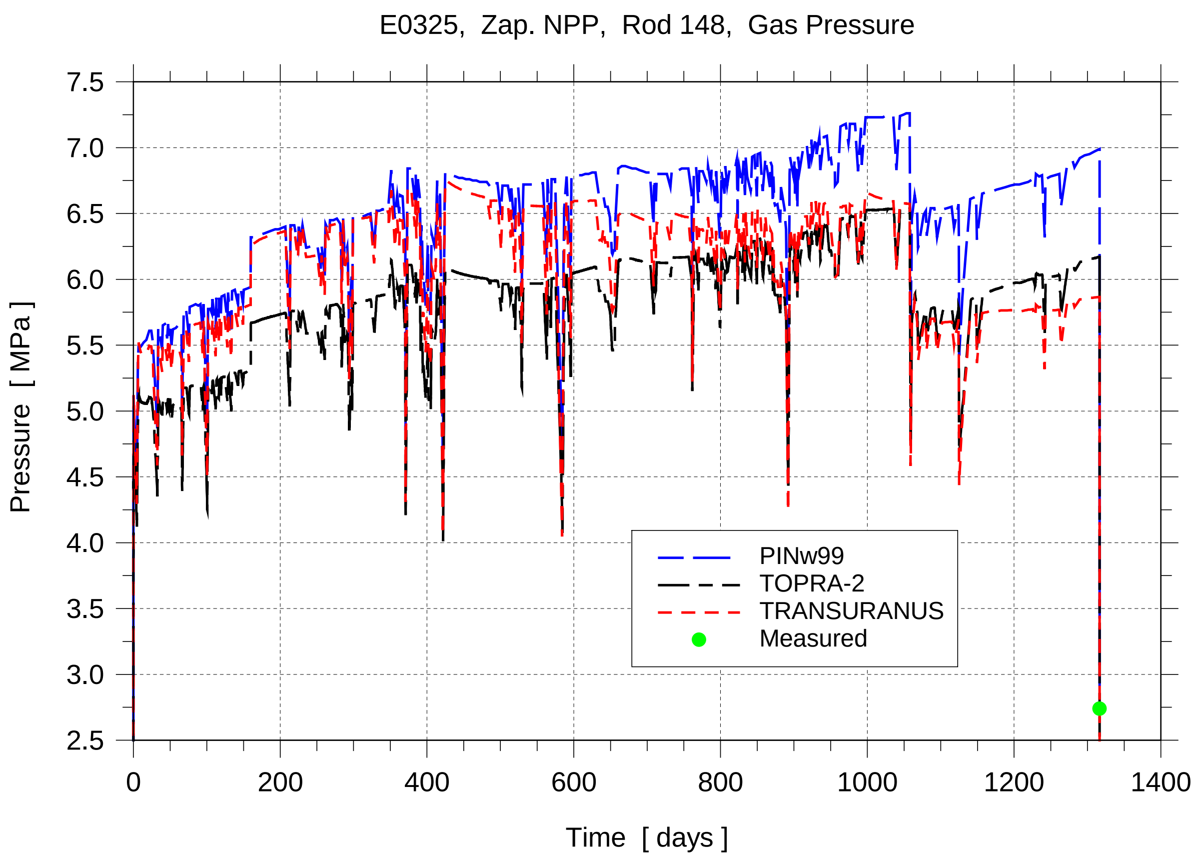


**Fig. 4.** Rod No.148 inner gas pressure

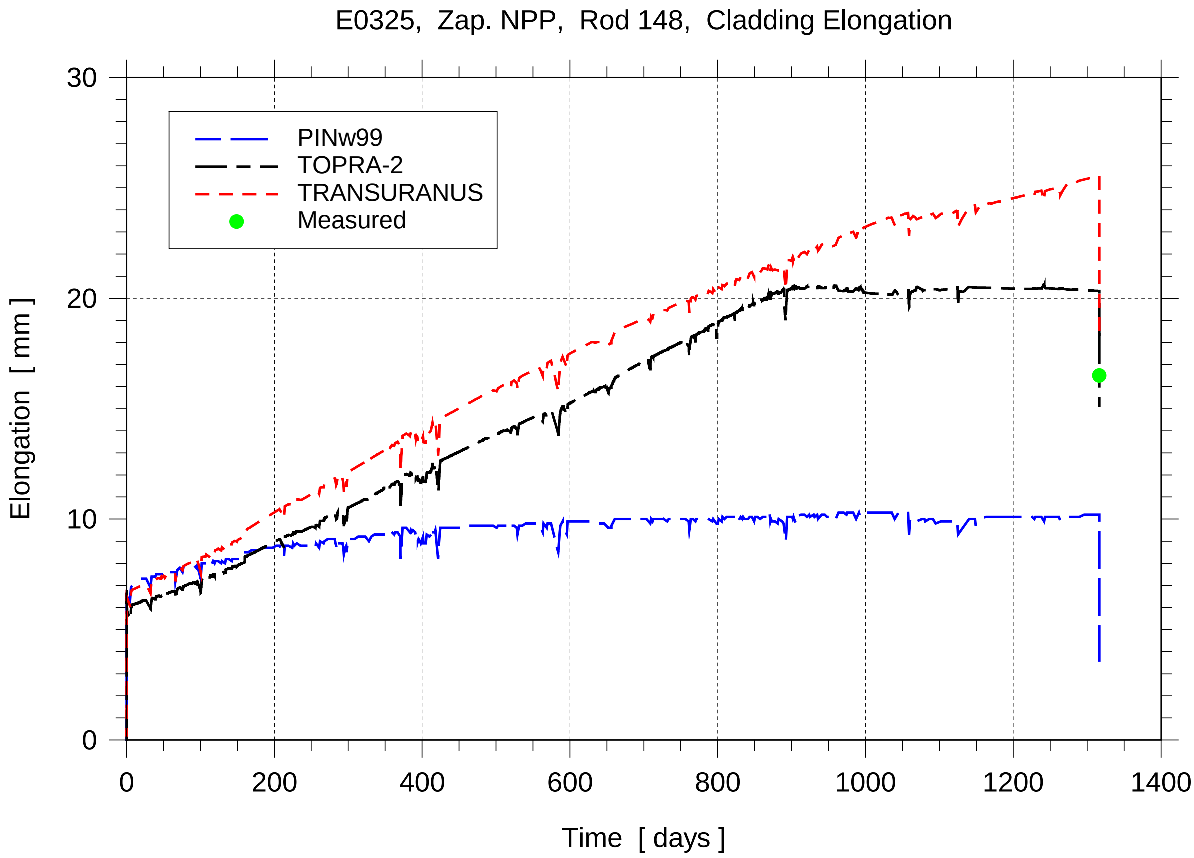


**Fig. 5.** Rod No.148 cladding elongation

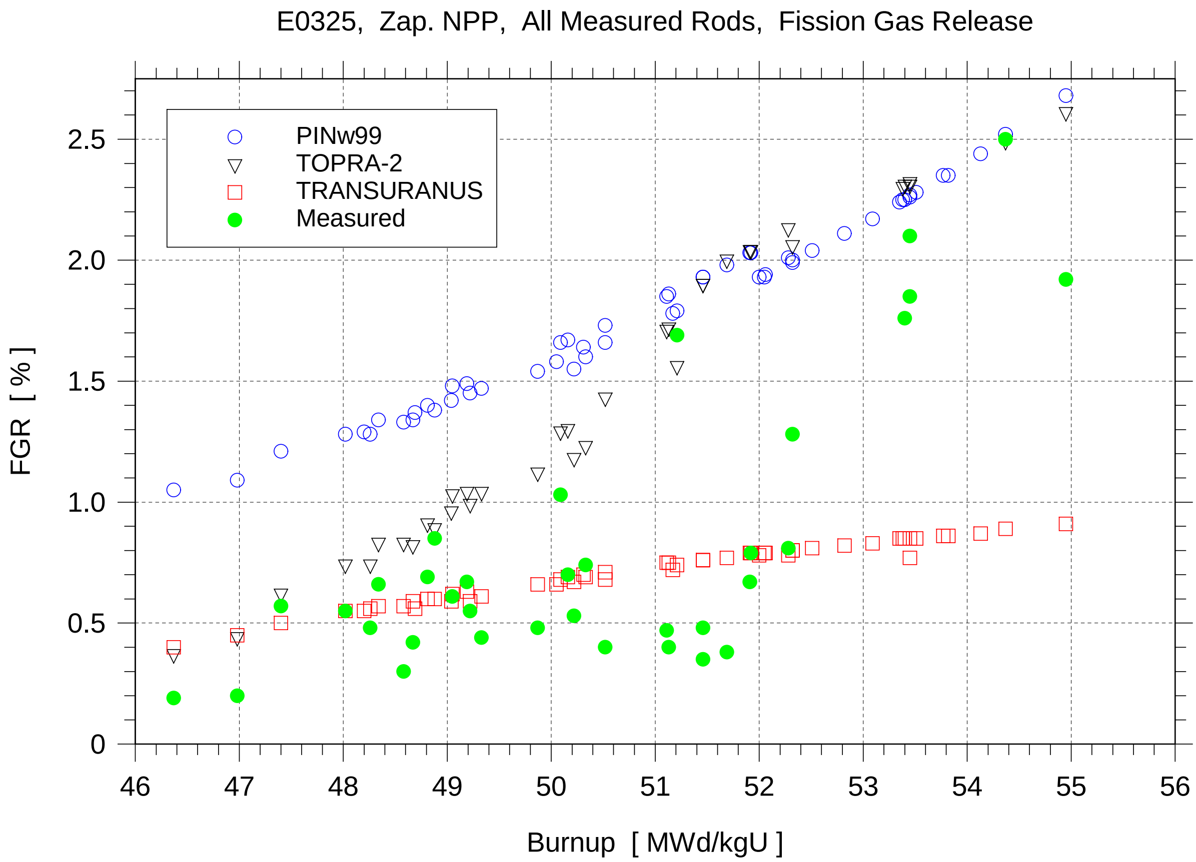


**Fig. 6.** Comparison of the calculated and measured FGR

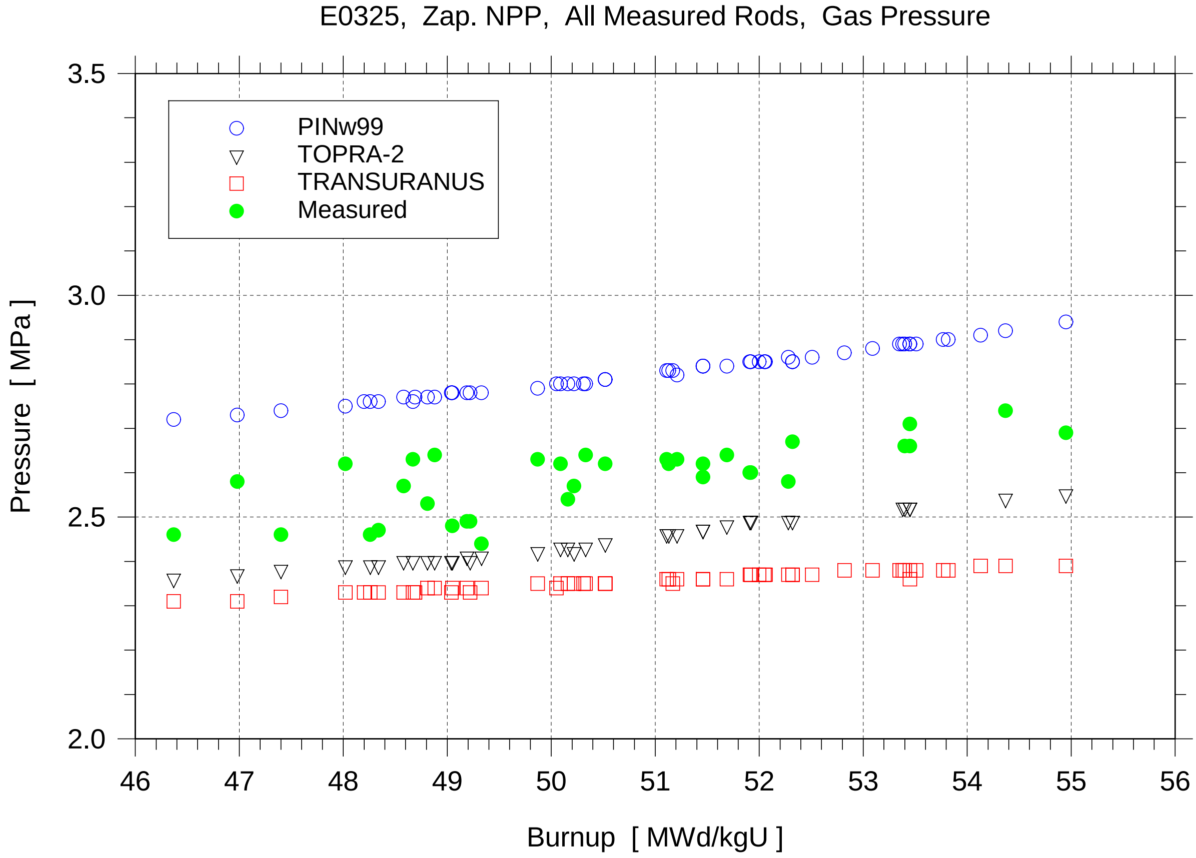


**Fig. 7.** Comparison of the calculated and measured inner gas pressure

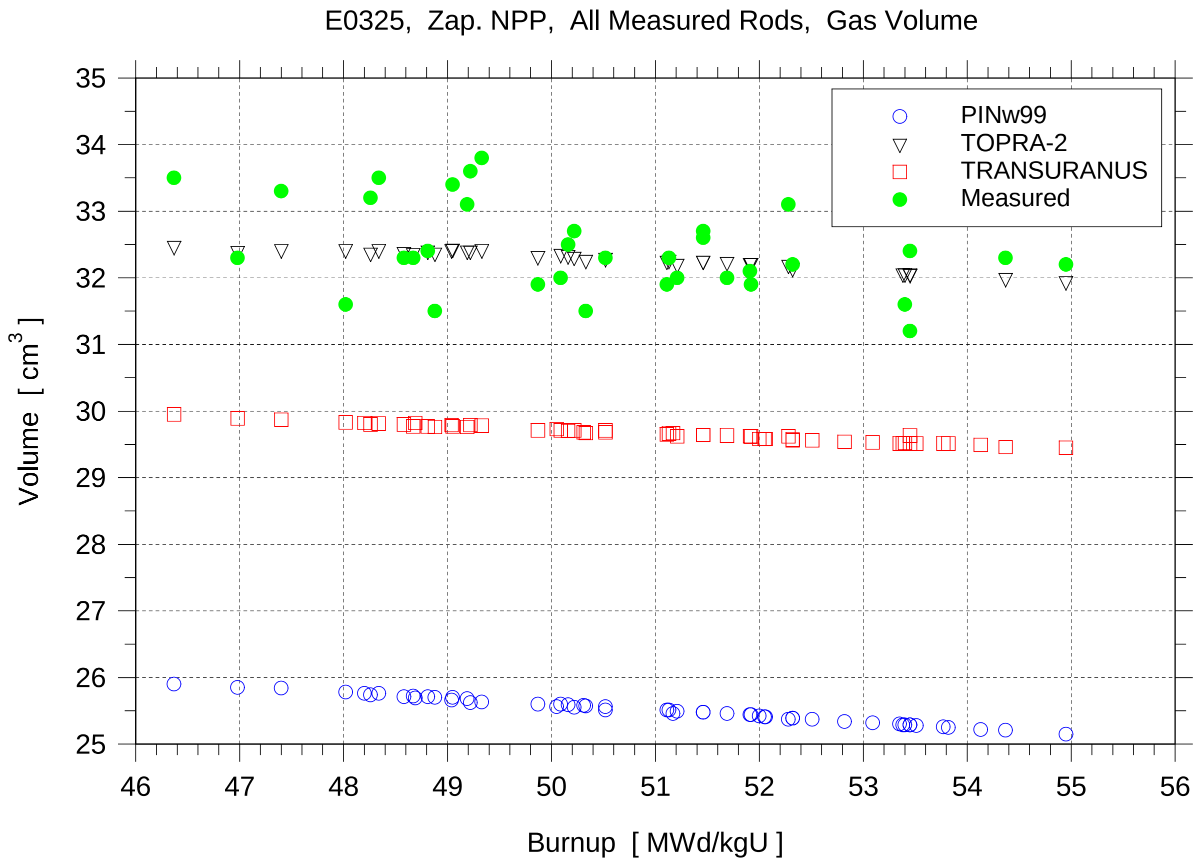


**Fig. 8.** Comparison of the calculated and measured inner gas volume

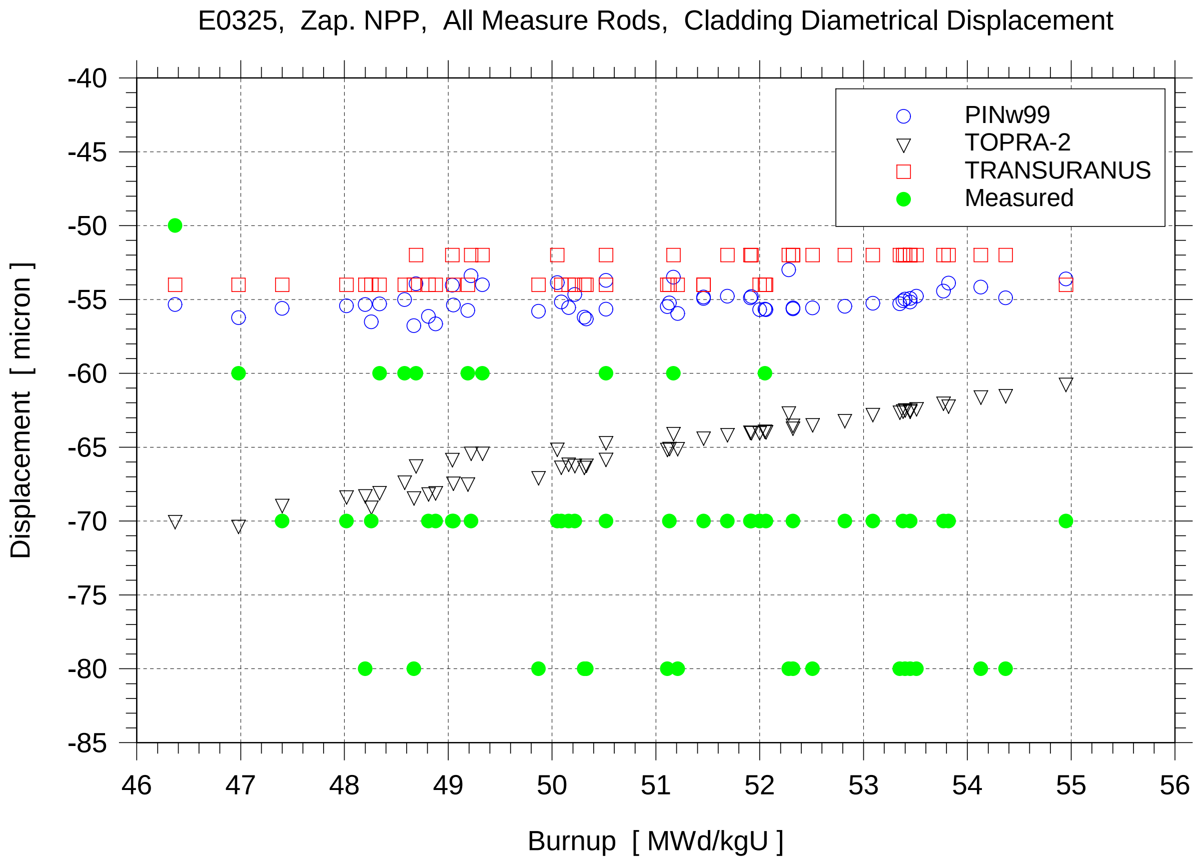


**Fig. 9.** Comparison of the calculated and measured cladding diameter change

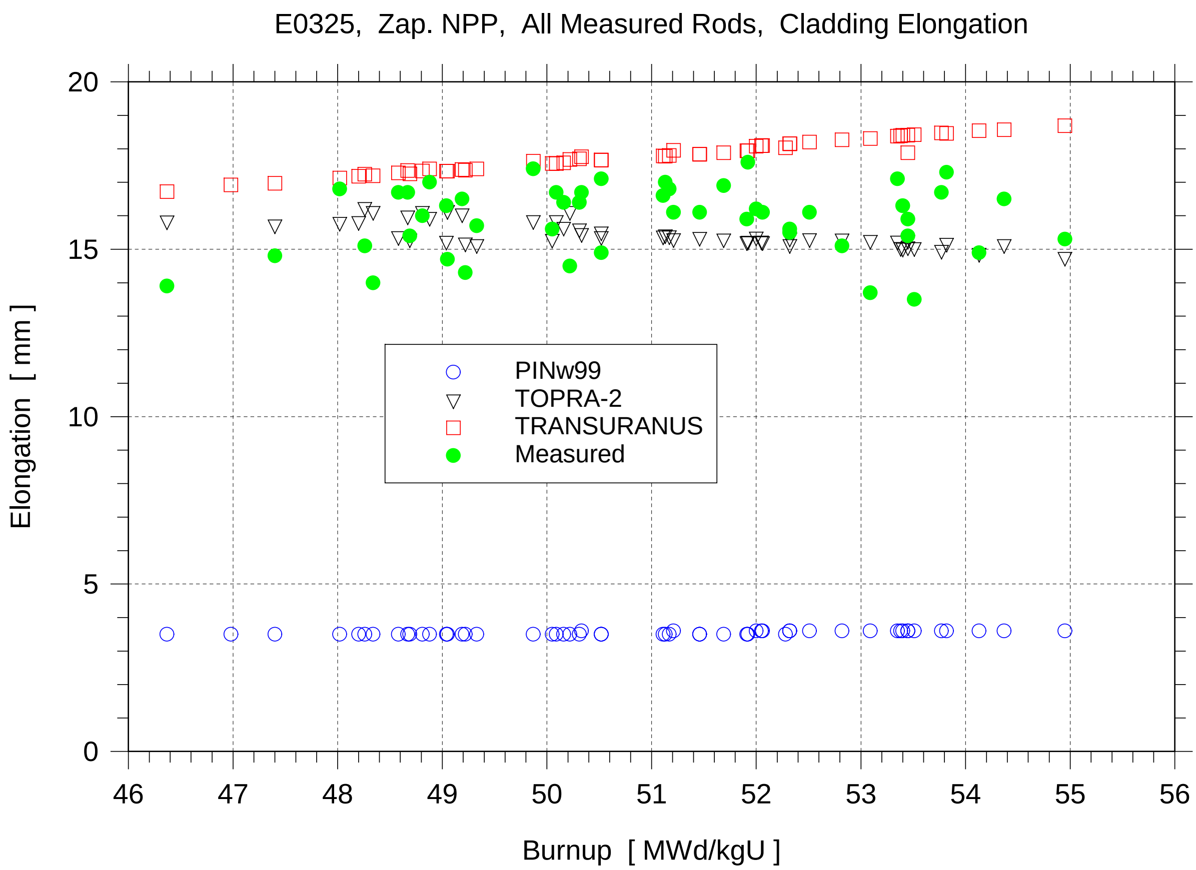


**Fig. 10.** Comparison of the calculated and measured cladding elongation