APPLICATION OF THE POOLSIDE FUEL INSPECTION RESULTS IN THE VALIDATION OF STATISTICAL FUEL ROD PERFORMANCE ANALYSIS

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Abstract: ÚJV Řež has been performing the analysis of the poolside fuel inspections at Temelin NPP since 2011. Although the main goals of the inspections were the qualification of new fuel design and root cause analysis in case of the presence of the leaking fuel in the core, significant amount of the information about the fuel behavior may be extracted from the obtained results and used in fuel rod performance code development and validation.
Current methodologies of the fuel performance calculations applied in ÚJV consider the uncertainties in fuel rod parameters and key code models using the statistical best estimate plus uncertainty (BEPU) approach. We have recently performed validation of this approach for the predictions of the fuel centerline temperatures using experiments performed in the Halden reactor, but the data for the validation of fuel rod deformation are much scarcer. This paper presents the application of the poolside measurement of the fuel rod elongation in the validation of the statistical approach to the fuel performance analysis.

Keywords: Fuel performance, Fuel Inspections, Uncertainties, Code Validation

INTRODUCTION

One of the tasks of the ÚJV Řež is the engineering and scientific support to the Czech utility ČEZ in the field of the nuclear fuel performance. The range of the provided analytical services is broad, from the core design calculations, thermal-hydraulics up to the CFD calculations, noise diagnostics to the fuel performance modelling. The TRANSURANUS code ([1]) supplemented by custom 2D and 3D FEM models in ABAQUS ([2]) is used to perform the fuel rod analysis. Apart from the analytical services, the personnel of UJV Group (UJV Řež and Centrum Výzkumu Řež) have performed poolside fuel inspections at Temelín NPP (2 units of VVER-1000) and have evaluated their results since 2011.

Although the main goals of the inspections were the qualification of new fuel design and root cause analysis in case of the presence of the leaking fuel in the core, significant amount of the information about the fuel behavior may be extracted from the obtained results and used in fuel rod performance code development and validation.

In the recent years, a small number of leaking fuel assemblies was revealed at Temelín units ([3]). Systematic investigation of the possible root cause was initiated. Since all the leaking assemblies were in the burnup range which is characteristic for the onset of the pellet-cladding contact, the capability of the fuel performance code used in the core design process to predict the pellet-cladding mechanical interaction was also questioned. It should be noted that pellet-cladding interaction was not suspected as a root cause due to excellent performance of the fuel assemblies with the equivalent fuel rod design at other plants and robust core design and operation process which eliminate any significant local power changes.

Validation of the fuel performance code should be done using the same methodology as the core design calculations, especially the treatment of the uncertainties. Our current methodology (Best Estimate Plus Uncertainty) is the following:

- Uncertainties of the linear heat rate are always added to the calculated value
- Uncertainties in the fuel rod as fabricated parameters are treated statistically, correlations between parameters and limiting fabrication tolerances (e.g. tolerances on fuel mass in fuel rod, which puts restrictions on the possible combinations of the fuel density and fuel pellet dimensions) are respected
- Uncertainties in the code models are treated statistically

The statistical analysis involves running *n* calculations with the uncertain parameters randomly sampled within fabrication tolerances / code model uncertainty range. The number of the runs is based on Wilk's formula, but increased over the number of the trials required for the conventional 95/95 confidence level (e.g. where 58 trials are needed for one-sided 95/95 bound we use 100 trials). We have recently performed validation of this approach for the predictions of the fuel centerline temperatures using experiments performed in the Halden reactor ([4]), but the data for the validation of fuel rod deformation are much scarcer.

POOLSIDE OBSERVATIONS

The elongation of fuel rods is routinely evaluated during the poolside inspections at Temelín using a combination of camera-positioning system information and image processing. This is possible due to the fact that the rods are mounted in the lower plate of the fuel assembly and fuel rods grow only upwards. The elongation is quantified for all rods in 1st and 2nd outermost rows. Two of the fuel assemblies inspected in 2017 have shown rod elongation which appeared counterintuitive. Fuel rods in the peripheral row of fuel assembly A appeared shorter than all interior rod in the same assembly (see Figure 1, left). Contrary, peripheral rods of B appeared longer than all the internal rods of the same assembly (see Figure 1, right). Both fuel assemblies were twice burned with similar power histories and burnup (26 – 27 MWd/kgU). Both assemblies were radially profiled – the peripheral row contained fuel rods with enrichment reduced with respect to the internal rods in order to compensate for increased moderation, but the mean enrichment of assemblies A and B was different.

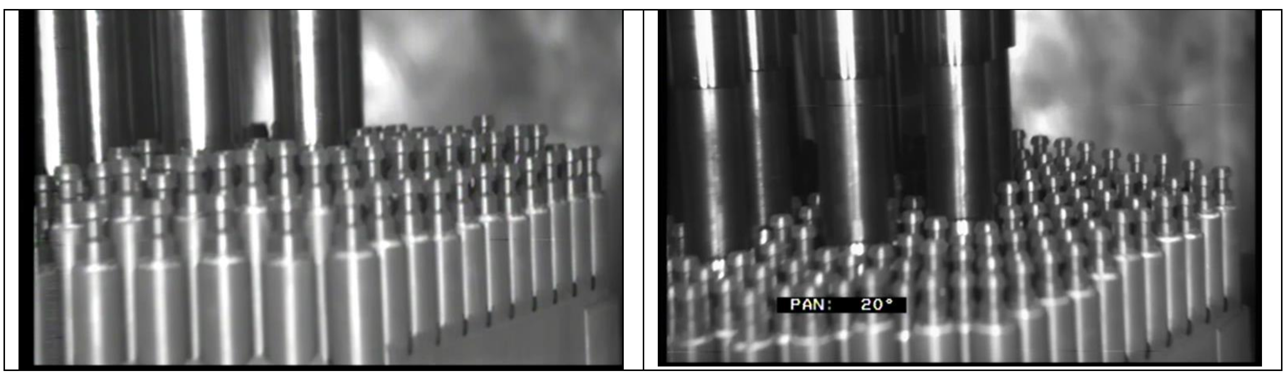

Figure 1. View of the top end plugs of fuel assembly A (left) and B (right).

Firstly, the rod elongation was quantified. Figure 2 shows the histogram of the rod elongation of A with respect to a reference rod (selected rod of the peripheral row for which the absolute elongation is determined by the camera positioning system). There was negligible burnup difference between these rods (up to 2 MWd/kgU) according to the core physics calculations. The results show two important facts:

- The average difference between the peripheral row and second row is 5 mm
- The spread of the results is much lower in the peripheral row, where the standard deviation (0.8 mm) is close to precision of the image processing (0.5 mm). In the second row, the standard deviation is 1.4 mm, with the elongation minimum values within the range of the 1st row and maximum difference between the 1st and 2nd row 7 mm.

The absolute elongation of the reference rod in peripheral row was evaluated to be 8.0 ± 1.5 mm. This is the most common value observed at Temelin current fuel design (TVSA-T) at this burnup.

Similar results were obtained on fuel assembly B, but with the internal rods near the expected value and the peripheral rods longer.

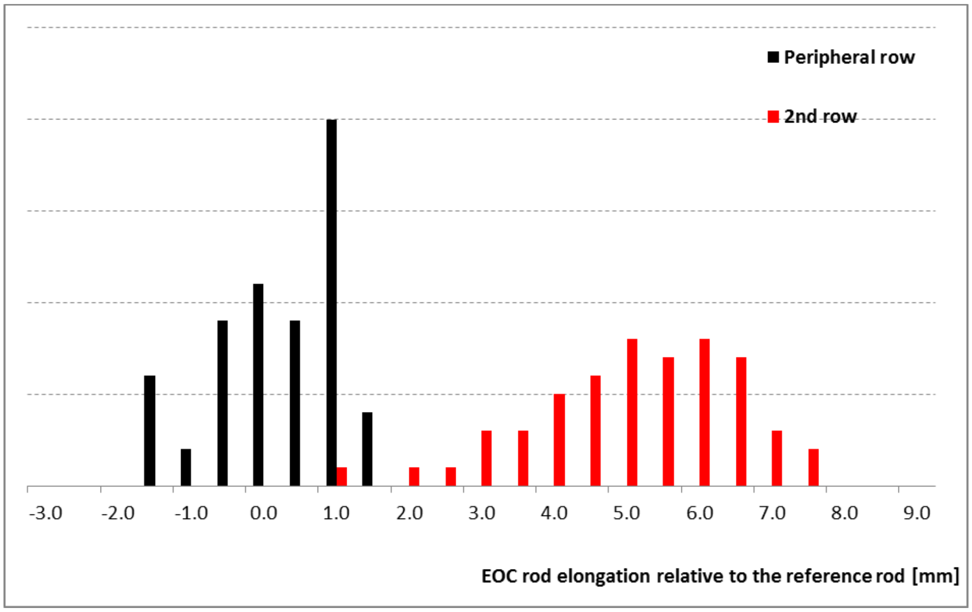

Figure 2. Histogram of measured elongation of the 1st and 2nd rows of fuel rods in fuel assembly A, relative to the reference rod elongation

Such difference between neighboring rods in the same assembly seemed unusual at first glance and therefore it was asked by the operator what the cause is and if UJV fuel rod performance calculations can reproduce this difference.

The first possible explanation was related to the power distribution. Fuel assembly deformation leads to the deviation in the water gap size from the nominal value. The rods in the peripheral row are most affected. However, this explanation was dismissed based on following observations:

- In fuel assembly A, the elongation of the peripheral rods is close the most common value of the fuel rod elongation in our database when the nominal burnup is considered, but the internal rods are at the upper bound of the database. If the difference was caused by the reduction of the power of the peripheral rods due to smaller water gaps, the internal rods should behave "normally" and the outer ones should be "shorter"
- The peripheral rods of fuel assembly A are shorter than the internal rods at all faces. We have checked the growth of the fuel rods at the neighboring fuel assemblies. If the reduction were caused by the power suppression in the peripheral row due to reduced water gap, it would be evident also on at least one of the neighboring assemblies on the face adjacent to the fuel assembly A – but none exhibited such behavior.

We have therefore concluded that the power distribution in the FA is not the primary cause of the observed difference. It should be noted that the uncertainties in the power distribution due to fuel assembly deformation are taken into account in a conservative manner during the core design.

The second possible explanation was related to the rod as manufactured properties. No common factor was identified for the cladding tubes, but the following observation was made for the fuel pellets:

- Both A and B were using rods with two different enrichments
- In total, three batches of fuel pellets were used in fuel assemblies A and B (excluding the gadolinia bearing rods, none of which is located in the 1st or 2nd row)
- One of these batches was used in all the internal rods of A and all the peripheral rods of B – i.e. rods with the elongation at the upper bound of the experimental database.

Obviously, all the pellets were manufactured within the prescribed tolerances. However, the tolerance interval is usually much broader than the range of the values within one batch. We have therefore decided to test, whether the fuel performance modelling can explain the observed difference in the fuel rod elongation based on the assumption that it is caused by the difference in the pellet properties within the fabrication tolerances.

MODELLING

Power histories of the fuel rods were reconstructed using the UJV core physics code ANDREA ([5]) from plant data. About 100 states were modelled in the history of each cycle, respecting all significant changes in the core power, axial offset and the positions of the control rods.

The TRANSURANUS fuel performance code was used for the modelling in the version v1m3j12 adapted at UJV (the most notable modification is the implementation of E110 alloy creep model according to [6]).

A statistical analysis was performed:

- One power history was modelled – a peripheral rod of fuel assembly A. It was confirmed that the difference in the calculated power history of this peripheral rod and neighboring rods of the second row is negligible. No uncertainty was considered in the power history for following reasons
  - we were interested in the relative difference between the peripheral rod and rod in a second row
  - apart from the possible effect of the inter-assembly water gap, these rods must have experienced almost the same power history (see above for the discussion of the water gap effect)
  - we appreciate that the uncertainty in the power determination is a significant factor. Therefore subsequent sensitivity analysis was performed. In this sensitivity analysis, the power was changed from the one used in the nominal case, but in the same manner for all the statistical runs to respect the fact, that the observed neighouring rods had pratctically same power history (the deviation from the nominal power history must have been the same for the peripheral rod and for the neighbouring rod in the second row)
- The uncertainties in the code models (fuel-cladding gap conductance, fission gas release…..) were not considered apart from several pellet specific models (see next point) because although there is some uncertainty in these models, it does not affect the relative behavior of the rods in the calculated case.
- Uncertainties related to the fuel pellets were considered in the statistical analysis, including
  - dimensions according to vendors specifications, uniform distribution
  - as fabricated porosity according to vendors specifications, uniform distribution
  - densification model according to vendors specifications (resintering) and Halden data (densification is assumed to be completed between 5 – 12 MWd/tU), uniform distribution
  - swelling model, normal distribution with a 5% σ applied to TRANSURANUS standard model
  - pellet-cladding friction coefficient, uniform distribution between 0.05 and 0.3
  - fuel relocation model, correction factor between 0.5 and 1.0 applied to the TRANSURANUS standard model
- Uncertainty of the fuel thermal conductivity model was not explicitly considered, but it is partially included in the calculation through the variation in the assumed pellet as fabricated porosity

The power history and predicted development of the fuel rod elongation is shown in Figure 3. In total, 500 calculations were performed. The minimum obtained rod elongation was 8.6 mm, mean 9.4 mm and maximum 11.8 mm. The difference between the predicted minimum and maximum value is 3.2 mm, which falls short of the observed mean difference 5 mm.

So far we have only considered the nominal power history. In order to see the effect of the uncertainties in the power, we have repeated the extreme runs (with minimum and maximum predicted elongation) with the power increased by a factor 1.04 in both cases (this is much less than the uncertainty conservatively assumed in core design). The minimum predicted elongation increased from 8.6 to 8.8 mm, the maximum one from 11.8 to 13.2 mm. The difference between the minimum and maximum increased to 4.4 mm which within the measurement error from the observed value.

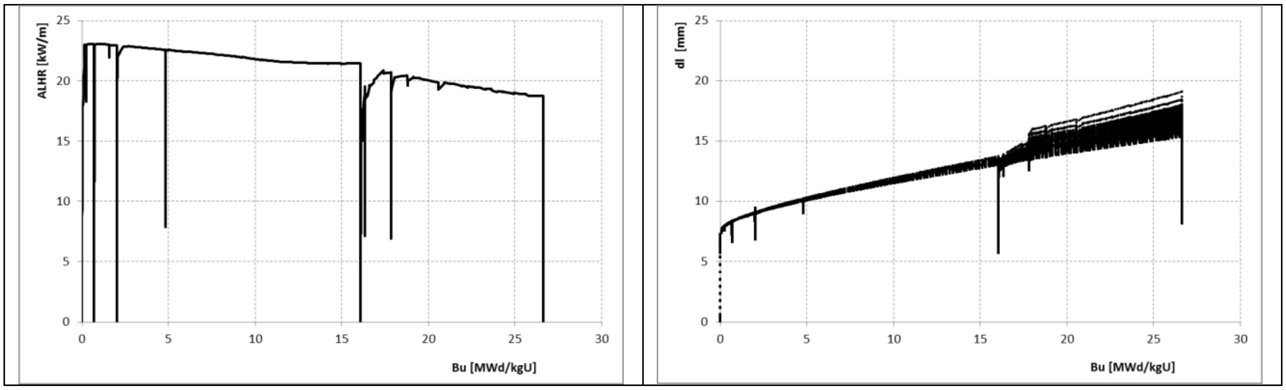

Figure 3. Power history used in modelling (left) and predicted fuel rod elongation (right)

A closer look at the obtained calculation results reveals the following facts:

- According the TRANSURANUS calculation, the origin of the spread of the rod elongations lies in the predicted fuel rod behavior during the ascension to the full power in the second cycle
- The changes of power during the start of the second cycle result in varying elongation, depending on the friction coefficient assumed in the calculation and the distribution of fuel-cladding gap width along the rod length (in our calculations, we allow for limited axial variations in as fabricated pellet dimensions and porosity)
- The predicted elongation is strongly correlated with the multiplication factor applied on the fuel relocation model and also with the assumed friction coefficients.
- In order to capture the observed difference in the elongations of the fuel rods, it was necessary to reproduce the power history in sufficient detail and not omit any significant power changes.

CONCLUSIONS

A statistical (Best Estimate Plus Uncertainty) approach to the fuel rod performance calculation applied at UJV in the core design analysis of Czech NPPs was tested against the poolside inspection observations. During the inspections at Temelin NPP a difference of 5 mm was observed between two groups of fuel rods in two twice burned fuel assemblies with the same burnup and power histories. While the maxima were within the expected range of the fuel rod growth, the difference observed within one assembly seemed unexpected and it was therefore asked if the fuel rod performance tools are able to explain it.

The calculations using the statistical approach were able to reproduce such differences based on the assumption that only parameters varying from rod to rod where those related to the fuel pellets. This result gives confidence in the prediction capabilities of the code with respect to the modelling of the pellet-cladding interactions. If the code and methodology were not able to reproduce the behavior of the fuel observed at the plant, it would not be reliable for example in the investigation of the possible root cause of fuel failures.

It should be also noted that the spread of the observed elongation was observed between nominally identical rods irradiated under identical conditions – this must be taken into account when the experiments are planned in the material test reactors. Due to decreasing number of available test facilities, there is a trend to replace the quantity of data with “advanced modelling”. However, if the basic process being studied is stochastic in nature, the modelling must also be based on a statistical approach and in order to validate it, and sufficient quantity of experimental data must be available.

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REFERENCES

[1] K. Lassmann. TRANSURANUS: a fuel rod analysis code ready for use, Journal of Nuclear Materials, 188, 295-302 (1992).

[2] ABAQUS 6.12. 2012. SIMULIA Dassault Systemes.

[3] D. Ernst, L. Milisdörfer. 5 years of experience with TVEL fuel at NPP Temelín. 11th International Conference on WWER Fuel Performance, modelling and Support. 2015, Varna, Bulgaria.

[4] J. Klouzal, M. Dostál, V. Matocha. Treatment of the uncertainties in fuel rod design analysis. Proceedings of TopFuel 2016. September 2016, Boise, Idaho, USA.

[5] F. Havlůj, R. Vočka, J. Vysoudil. “Andrea 2 – Improved version of code for reactor core analysis”, in proceedings of the 22nd International Conference on Nuclear Engineering (ICONE22), July 7-11, 2014, Prague, Czech Republic.

[6] A.Ya. Rogozyanov, G.P. Kobylyansky, A.A. Nuzhdov. Behaviour and Mechanisms of Irradiatio-Thermal Creep of Cladding Tubes Made of Zirconium Alloys, Journal of ASTM Int., Vol 5 No 2, 2008.