Predictive capabilities of the TRANSURANUS fuel performance code for Russian-type WWER reactor

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Abstract

For commercial WWER, operated in the East European countries, one of the most powerful computer codes for assessing fuel performance is TRANSURANUS, developed by K.Lassmann and collaborators [1] to high level of reliability at the European Commission Institute for Transuranium Elements in Karlsruhe, Germany. The code offers a rich spectrum of possibilities to predict the physical status of the fuel at different moments of the fuel cycle and, on this basis, for decision making.

The code is able, starting from the manufacturing data of the operated fuel, reactor and fuel-cladding material properties and operational history of the power reactor, to make prediction on the fuel behaviour characteristics for more than 30 parameters, for both steady-state and transient regimes. The code has been verified and validated over numerous experimental data, mainly those, distributed by OECD-NEA/IAEA and known as IFPE (International Fuel Performance Experiments) Data base[2]

Introduction

The TRANSURANUS application in the case of WWER reactors requires to adopt and to verify the code to the specific design characteristics of both fuel and cladding materials used in WWER reactors. As the annular cylindrical geometry of the WWER fuel was already treated as a standard option in the TRANSURANUS code the most important specific models for the WWER version of the code are related to the Nb-containing cladding behaviour. Most of the relevant mechanical and thermal properties of the Zr1%Nb cladding were directly derived from the open literature.

During the last ten years, the WWER version of the TRANSURANUS code was verified against Russian experimental data. The comparison of the code predictions with the experimental data showed high-level “calculation-experiment” compatibility, so the code is fully applicable to operational practice of acting NPP’s, based on WWER-type of nuclear reactors[3].

Verification by the IFPE data base

Basic verification work has been performed using experimental data of nuclear fuel irradiated in WWER-440 reactors - Sofit and Kola-3 experiments. The activities were broadened towards - WWER-1000 fuel, first post-irradiation computations based on new data prepared for the IFPE database (Zaparozhje and Novoronezh experiments).

Data of specific interest for WWER applications included in IFPE data base:

a) 12 rods from the Finnish-Russian program SOFIT. The data set available to date contains in-pile measurements of fuel centre temperatures and in-pile fuel and clad extension measurements.
b) Kola3 experiment includes WWER-440 fuel rods irradiated during 4 cycles (FA198) and 5 cycles (FA222) under normal operational conditions in the KOLA NPP, Unit 3. Maximum burnup reached is up to ~ 50 and 60 MWd/kgU. Power histories and consequently post irradiation examinations (PIE) for 32 rods are available in the IFPE database. PIE observations cover dimensional cladding and fuel changes and FGR.

c) Zaporozskaya WWER-1000 experiment. FA0325 with well defined fuel and cladding properties was irradiated up to assembly averaged burn-up of 48.9 MWd/kgU and PIE data on rod length, clad diameter changes, fuel-to cladding residual gap and fission gas release are ensured.

d) Novovoronezh WWER-1000 experiment. FA4108 was operated in the fifth unit of Novovoronezh NPP during 3 fuel cycles to an average fuel burn up of 44 MWd/kgU. After irradiation, the assembly was shipped to the hot cells of NIIAR, Dimitrovgrad where the rod length, cladding diameter and fuel-to cladding gap were measured.

Results of the verification

The TRANSURANUS-WWER code verification for different fuel behaviour characteristics performed at INRNE in close collaboration with ITU Karlsruhe (Germany) showed that the calculations give very satisfactory predictions of a) the fuel behaviour during operation and b) the state of the fuel after irradiation.

Fuel Temperature

The SOFIT 1.1 is a test WWER fuel assembly consisting of 18 fuel rods with varying design parameters. Six of the rods were instrumented by centerline thermocouples and the database contains experimental data on the in-pile measured fuel central temperature. The rods irradiated in the frame of the SOFIT program, were of different initial gap size (~ 150, ~210, ~270µm) and fill gas pressure (0.1, 0.5 amd 1.5 MPa). The fuel pellet density was typical for sintered WWER fuel ~96-97 % of the theoretical density. The rods reached average burnups 8-10 MWd/kgU.

The coincidence experimental-calculated temperatures for all 6 rods is shown in Fig.1. The calculated temperatures are in very good agreement with the measured ones. The general conclusion, following from the analysis above, is that the TRAMSURANUS predicts very precisely the FCT in the Sofit-1.1 rods.

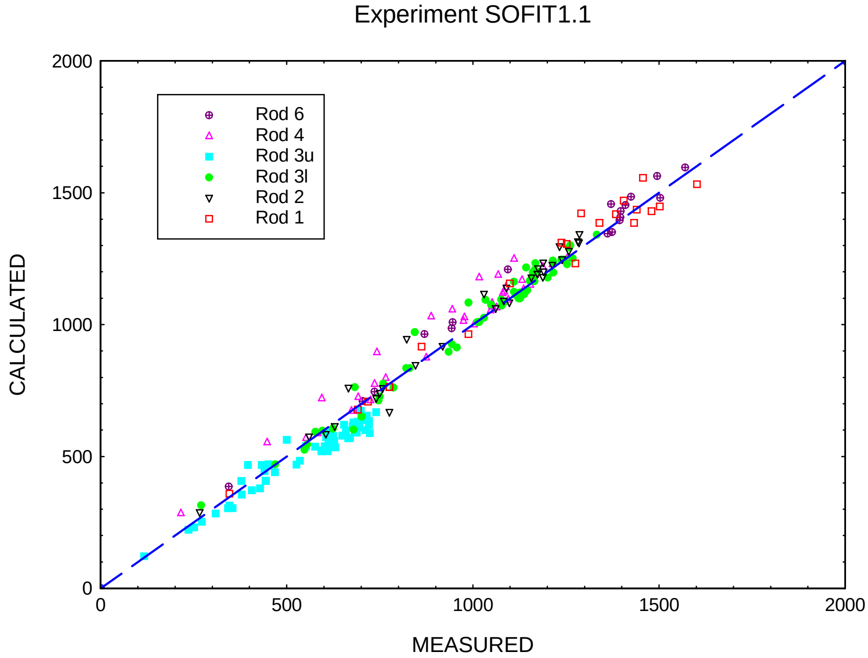

Fig.1. Comparison of the measured Fuel Central Temperature and TRANSURANUS code predictions for the 6 instrumented rods from the SOFIT1.1 experiment

Fission gas release

The measured and calculated fission gas release in the 22 WWER fuel rods for wich PIE data became available are shown in the

Fig.2. The data indicate an enhanced athermal gas release above rod average burnup of 40 MWd/kgU. This phenomenon is modeled through an optional threshold for the onset of the enhanced release. The threshold was expressed in terms of local burnup and was tuned on the bases of the Kola3 data set. With this option, however the code tends to under-predict the FGR in the high-burnup WWER-1000 rods.

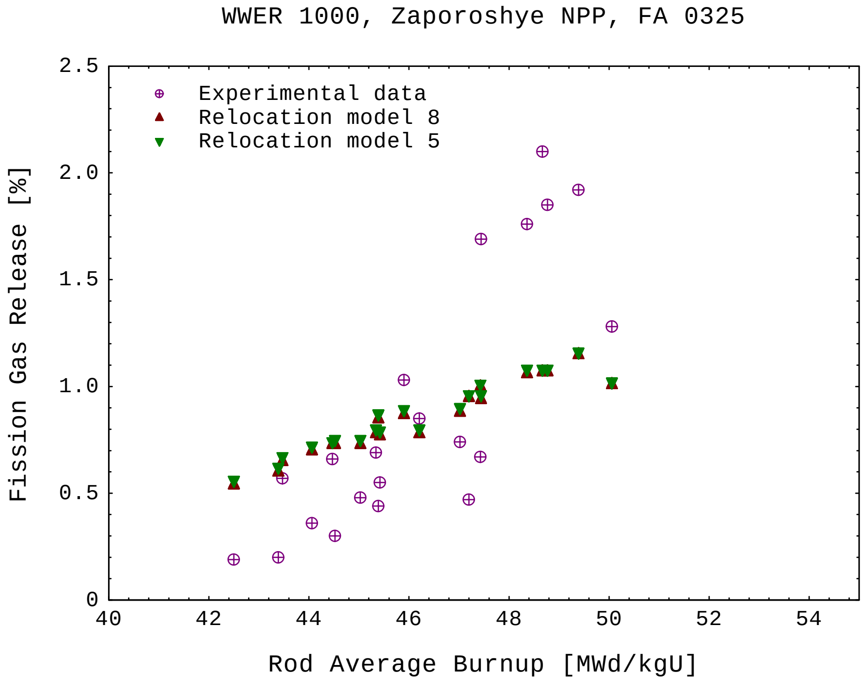

Fig.2. Comparison of fission gas release – TU code predictions and puncturing results from post irradiation examination

Change of Fuel Rod Diameter and Length

The residual geometrical changes of the fuel rod are mainly influenced by two phenomena: a) the creep-down of the cladding tube due to overpressure of the coolant; b) the axial growth of the cladding due to irradiation.

In

Fig. 3 the change of the fuel rod diameter predicted by TRANSURANUS at EOL is compared with the measured values for one rod from FA222 irradiated in the Kola3 NPP during 5 cycles. The dots mark the experimental values and the lines represent the TRANSURANUS calculation.

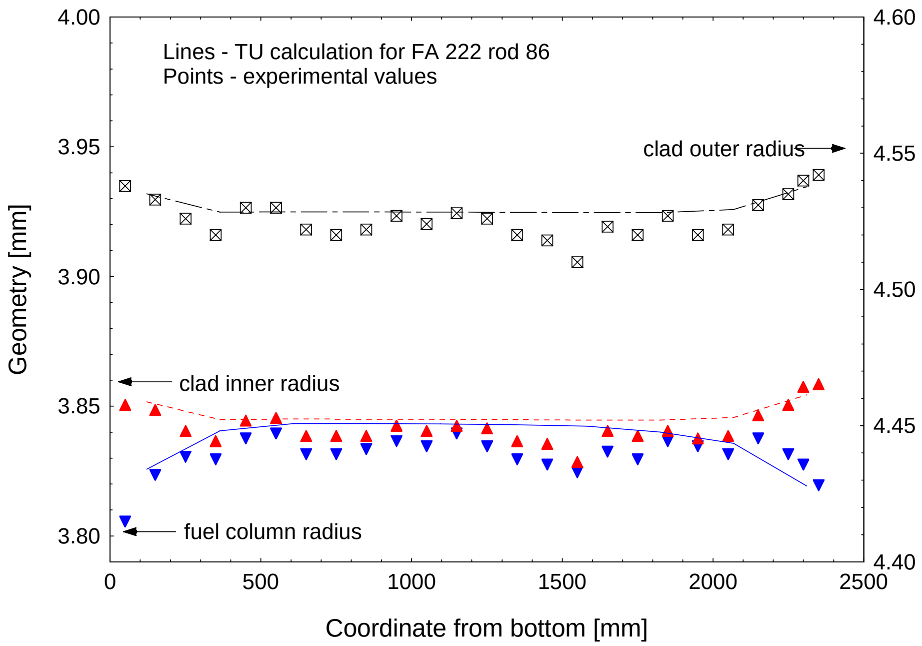

Fig. 3. Axial profile of cladding outer and inner radius and fuel outer radius for rod 86 (Kola3 experiment) calculated with standard TRANSURANUS options, compared to the experimental values.

The results of calculated/measured comparison in the case the WWER-1000 fuel rods can be regarded as satisfactory, although the TU simulation of the FA0325 produced slight over-prediction of the rod axial elongation. In the Fig. 4 the results for axial elongation and the fuel-to-cladding gap of the FA4108 TU simulation are presented.

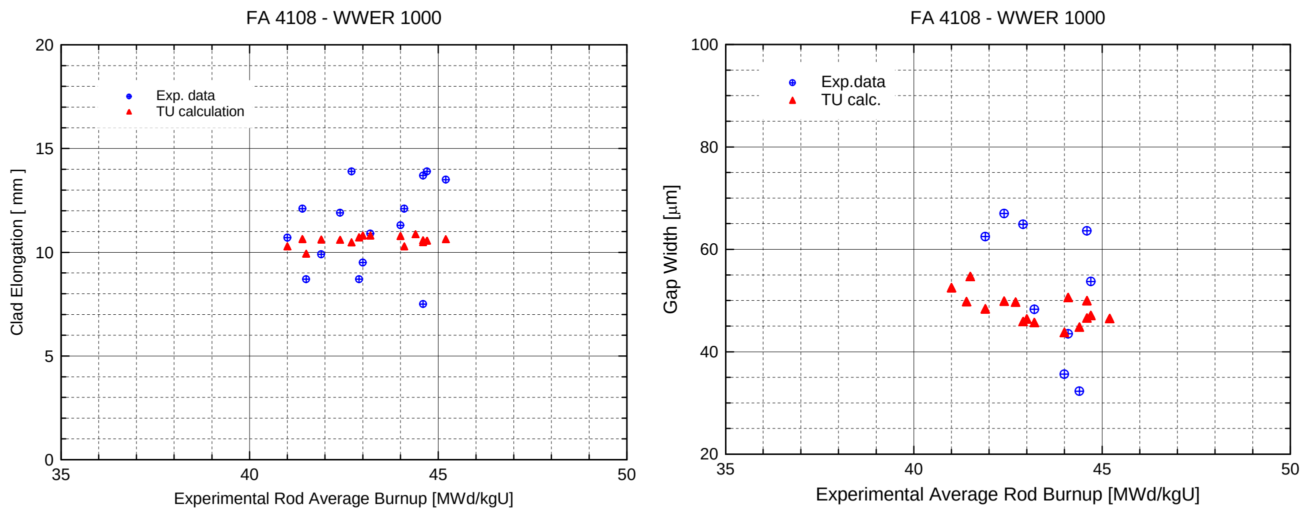

Fig. 4..WWER-1000 Novovoronezh NNP experiment. Comparison of the gap width and cladding elongation, measured and calculated as a function of the rod burnup.

The Bulgarian NPP Kozloduy runs two types of WWER units – WWER-440 and WWER-1000, where the numbers 440 and 1000 denote the electrical power of the WWER-type reactors. Both types of reactors use standard fuel, operated in all WWER units in Eastern Europe. This is the reason, for which the TRANSURANUS code was implemented at the NPP Kozloduy.

For both WWER fuel types the analyses were complemented by the simulation of operational irradiation histories of the Kozloduy NPP. Calculations of the fuel pins from the NPP Kozloduy in Bulgaria have been done for the first time by the TRANSURANUS code. So far no in-pile or post-irradiation experimental data of nuclear fuel used in Kozloduy are available for comparison with the TRANSURANUS calculations. For conditional check-up a comparison has been made with similar PIE data of standard WWER-440 and WWER-1000 fuel from the IFPE database

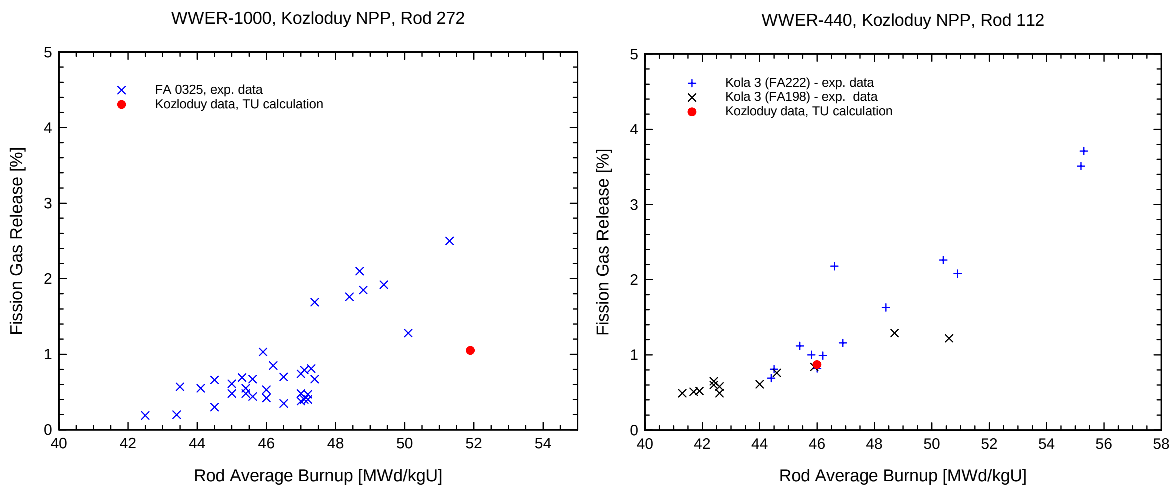

Fig 5. Release of fission gas vs. averaged rod burnup. Comparison is made to the experimental data for both reactor types.

The results at burnup up to 46 and 52 MWd/kgU respectively are in good coincidence with the experimental ones for similar fuel. This allows to use the code for decision-making in the NPP Kozloduy everyday practice.

As general conclusion to the analysis of the WWER nuclear fuel, for extending the verification, further WWER-1000 as well as WWER-440 fuel rods have to be simulated; including completely independent irradiation experiments. For example, the information available from the IFPE database, could be applied as more comprehensive original datasets of WWER fuel irradiated at the OECD Halden reactor.

Acknowledgement

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With Dancho Elenkov’s passing, the Group of Nuclear Fuel Modelling in INRNE has lost a highly distinguished and dedicated researcher and leader. He will be gratefully remembered by us his colleagues and friends.

References

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