Progress Report under IAEA Research Contract №15370/L2

Title of Project: *"Investigation of PWR and VVER Fuel Rod Performances under High Burnup Using FEMAXI & PAD Codes"*

Report

STEADY STATE AND TRANSIENT FUEL ROD PERFORMANCE ANALYSES BY PAD AND TRANSURANUS CODES

Host organization: "Nuclear Fuel Cycle" Science and Technology Establishment (NFC STE), National Science Center "Kharkov Institute of Physics and Technology (NSC KIPT)

Contractor:

Director, NFC STE NSC KIPT

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Reported period: April 14, 2011 – May 30, 2012

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Revision History

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0	Initial edition	O. Slyeptsov	May 2012

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List of abbreviation

ADU	Ammonium DiUranate (fuel pellet manufacturing process)
BE	Best Estimate
BOC	Beginning- Of-Cycle
BOL	Beginning-Of-Life
BU	Burnup
CRCD	Center for Reactor Core Design
E110	Zr-1%Nb alloy marketed by Russian corporation “TVEL”
FA	Fuel Assembly
FR	Fuel Rod
FGR	Fission Gas Release
GWD/MTU	Giga-Watt Day per Metric Tonne of initial Uranium metal
HBS	High Burnup Structure
IAEA	International Atomic Energy Agency
IDR	Integrated Dry Route powder conversion process
IFPE	International Fuel Performance Experiments
ITU	Institute for Transuranium Elements
LHR (LHGR)	Linear Heat Generated Rate
LWR	Light Water Reactor
MWD/MTU	Mega-Watt Day per Metric Tonne of initial Uranium metal
ND	Nominal Dimension
NEA	Nuclear Energy Agency
OECD	Organisation for Economic Co-operation and Development
PAD	Fuel rod design code developed by Westinghouse
RIP	Rod Internal Pressure
PWR	Pressurized Water Reactor
WWER	Russian-designed pressurized water reactor
TU	TRANSURANUS code developed by ITU
WEC	Westinghouse Electric Company
UNFQP	Ukraine Nuclear Fuel Qualification Programme
ZIRLO	Low-corrosion zirconium alloy marketed by Westinghouse

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Executive summary

The report performed under IAEA research contract №15370/L2 describes the analysis results of WWER and PWR fuel rod performance at steady state operation and transients by means of PAD and TRANSURANUS codes. The code TRANSURANUS v1m1j09 developed by Institute for of Transuranium Elements (ITU) was used based on the Licensing Agreement N31302. The code PAD 4.0 developed by Westinghouse Electric Company was utilized in the frame of the Ukraine Nuclear Fuel Qualification Project for safety substantiation for the use of Westinghouse fuel assemblies in the mixed core of WWER-1000 reactor.

The experimental data for the Russian fuel rod behavior obtained during the steady-state operation in the WWER-440 core of reactor Kola-3 and during the power transients in the core of MIR research reactor were taken from the IFPE database of the OECD/NEA [1, 2] and utilized for assessing the codes themselves during simulation of such properties as fuel burnup, fuel centerline temperature (FCT), fuel swelling, cladding strain, fission gas release (FGR) and rod internal pressure (RIP) in the rod burnup range of (41 – 60) GWD/MTU.

The experimental data of fuel behavior at steady-state operation during seven reactor cycles presented by AREVA for the standard PWR fuel rod design [3] were used to examine the code FGR model in the fuel burnup range of (37 – 81) GWD/MTU.

The fuel rod performances analysis performed using the TRANSURANUS code revealed the following:

- the calculated burn-ups, such as the rod average (<BU>) as well as the maximum (BU_{max}), are in satisfactory agreement with the measured ones. Thus, for the WWER-440 fuel rods (FRs) examined the deviation between the measured and code predicted burnup is - 0.2 % (Standard Deviation (S.D.) = 2.4 %) and 3.6 % (S.D. = 1.8 %) for the rod average and the maximum fuel burnup, respectively. For the PWR rod simulated, the deviation of <BU> at EOL is 3.2 %;
- for the simulated transient condition, the fuel centerline temperatures (FCTs) calculated for the WWER fuel are in satisfactory agreement with the experimental data obtained for the fuel with burnup of ~50 GWD/MTU. For the high-burnup (58.4 GWD/MTU) the FCTs recorded are (190 ÷ 300) ^{0}C higher than the calculated ones;
- for the steady state operation, the calculated FGR data are in satisfactory agreement with the experimental data for the WWER-440 FRs having annular pellets as well as for the PWR rods with the solid ones. The calculated RIP data follow the data of gas release. Thus the deviation of RIP (EOL, normal condition) averaged over all the WWER-440 FRs simulated is underestimated and is 0.28 MPa (S.D. = 0.13 MPa).

  For transient conditions, the FGR calculated for the refabricated WWER-440 rods with different initial burn-ups are significantly below the measured ones. At the rod power of (20÷25) kW/m the predicted RIP is 3.8 MPa and 7.5 MPa lower than the measured one for the rod with BU= 48.9 and 60.5 GWD/MTU, respectively;
- for the WWER-440 FRs operated at "normal" condition the calculated values of cladding outer diameter (DCO) decrease at EOL are slightly underestimated at the burnup ranged in (45 ÷ 55) GWD/MTU and are in good agreement with the measured ones, when the fuel burnup grows up to 60 GWD/MTU;
- for the examined WWER-440 rods the calculated data of pellet-to-cladding diametral gap change versus fuel burnup are in satisfactory agreement with the experimental data, when the modified *FRAPCON*-3 fuel relocation model is used.

The gap values measured for the refabricated FRs after the power ramp tests overpredict the calculated ones. In the range of the power ramp of (34 ÷ 45) kW/m the deviations between the measured gaps and the calculated ones is about 20 μm.

The results of fuel rod analysis obtained by PAD code show:

- for both fuel rod designs, WWER-440 and PWR, the predicted fuel burn-ups agree well with the experimental ones. The deviation between the measured and the calculated burnup averaged over all the WWER-440 FRs tested does not exceed 1.6 % (S.D. = 1.8 %) and -3.1 % (S.D. = 2.5 %) for the rod <BU> and BU_{max}, respectively;
- for the transient conditions, the FCTs calculated for the refabricated WWER-440 rod with a burnup of ~50 GWD/MTU are in satisfactory agreement with the measured ones and are underestimated for the rod with a burnup of 58.4 GWD/MTU. For the last one, the deviation of FCT varies in the range of (200 ÷ 300) ^{0}C, when the rod average power changes from 21.3 kW/m to 25.1 kW/m;
- the steady-state FGR data calculated for WWER-440 rods are in good agreement with the experimental data. For the PWR rod tested the predicted FGR vs. burnup falls within the measurement uncertainties and at the rod burnup of 81.5 GWD/MTU the FGR is 11.3 % versus the measured 9.0 (+2.5/-2.0) %;

  For the transient conditions, the FGR values calculated for the refabricated WWER-440 FRs are underestimated. At the initial FGR-model parameters and the model parameter accounting for the fuel thermal conductivity degradation with burnup, the deviation of FGR averaged over all rods examined is 10.7 % (S.D. = 8.8%). The deviation decreases to 6.9 %, when the modified FGR model parameters for the high-burnup range are used;
- the predicted RIPs at EOL for WWER-440 rods examined agree well with the measured ones. The deviation between the measured and the calculated RIP for most rods tested falls within the manufacturing tolerance of ±0.1 MPa.

  For the transient conditions, the prediction of RIP at power for the refabricated WWER-440 rods (48.9 and 60.5 GWD/MTU) is underestimated, when the original FGR model parameters are used. For the high-burnup fuel rod the underestimation is 4.3 MPa at the rod power of 19.5 kW/m and it decreases to 2.8 MPa, when the modified parameters of FGR model are used;
- the data of fuel volume change ($\Delta V/V_0$) versus burnup calculated using the “WEC” fuel densification and swelling model are in satisfactory agreement with the experimental data. In the case, when the “NRC” fuel swelling model with the densification rate of 1.2 % is used, the overestimation of $\Delta V/V_0$ is observed, which reaches ~1% at BU = 61 GWD/MTU;
- after the base irradiation of WWER-440 rods, the predicted DCO change at EOL agree well with the measured ones. The deviation between the measured and the calculated ΔDCO averaged over all rods examined is -2.5 μm and -3.5 μm (S.D. ≈ 10 μm) for the “NRC” and the “WEC” fuel swelling model, respectively;
- within the design tolerances for cladding and fuel pellet diameter of WWER-440 rod, the predicted pellet-to-clad gaps are in good agreement with the experimental data. This is observed for both NRC and WEC fuel densification and swelling models used.

  The calculated gaps after the power ramp correlate well with the measured ones and show the growth of gap with the increase of the linear heat rate (LHR). The diametral gap predicted after the ramp with the peak LHR of ~45 kW/m is 70 μm versus the measured 65 μm.

1. WWER-440 Fuel Rod Performance Analyses

1.1 WWER Fuel Rod Specification

The fuel rod (FR) component characteristics of Russian WWER-440 fuel used in the assemblies of FA-198 and FA-222 are taken from IFPE database [1, 2], as well as from [4, 5, 6, 7].

The main FR parameters used in the rod performance evaluation are presented in Table 1-1. The additional data of WWER-440 fuel were also utilized during the analysis:

- the pellet outer diameter (DP) of 7.59 mm [4]. As noted by the Russian designer in the report, this DP value is one of the main differences of WWER-440 pellet vs. WWER-1000 pellet of 7.55 mm;
- the data of WWER fuel densification vs. the initial density of pellets sintered at 1700 ^{0}C, which were presented in [8];
- the data of the cladding outer diameter (DCO) for the individual WWER-440 FRs, which were taken from the cladding profilograms presented by Russian investigators in [5].

1.2 WWER Operational Conditions and Post-Irradiation Examination

1.2.1 Base Irradiation

The assemblies FA-198 and FA-222 were operated in 4- and 5-year cycle regimes in the WWER-440 core of Kola NPP Unit 3 (Kola-3) up to the average burnup of 45.9 GWD/MTU and 49.3 GWD/MTU, respectively [1]. The thermal-hydraulic parameters of WWER-400 core were within the nominal operating conditions – the coolant inlet temperature is 265 ^{0}C; coolant pressure is 12.3 MPa.

During the operation the maximum LHR for the FA-198 FRs varied in the range of (182 ÷ 257) W/cm at the beginning of life (BOL), (137 ÷ 151) W/cm at the beginning of 4-th cycle (BOC-4) and (149 ÷ 156) W/cm at the end of life operation (EOL). For the FA-222 rods the maximum values of LHR were (82 ÷ 183) W/cm at the BOL, (177 ÷ 219) W/cm at the BOC-2, (108 ÷ 123) W/cm at the BOC-5 and (100 ÷ 154) W/cm at the EOL.

The power histories of 19 rods for FA-198 and 22 rods for FA-222 are available in IFPE database [1] and were used for the rod performance simulation.

After the base irradiation, the following examinations of FA-198 & 222 fuel rods were performed:

- the fuel burnup. The measured uncertainties for the rod average as well as the maximum fuel burnup were ± 7 %;
- the pellet-to-cladding diametral gap. In the gap range of (0 ÷ 110) μm the measured uncertainty was ± 10 μm [5]. The data of gap variation along the fuel stack and the gap averaged in the central part of the fuel rod are presented in IFPE database [1];
- the cladding outer diameter change. The measured uncertainty of DCO change along the fuel stack was ± 10 μm [5]. The data of ΔDCO averaged in the axial range of (30-85) % of the fuel stack length are presented in IFPE database [1];
- the RIP and fission gas release. The measured uncertainties are not presented for the rod examined;
- the fuel rod elongation. The ultimate measurement error of the fuel rods length was ± 0.3 mm.

Table 1-1. WWER-440 Fuel Rod Specifications of FA #198 and #222

Parameter	Value
Fuel Assembly and Fuel Rod	
Fuel assembly (FA) type	Hexagon
FA pitch, mm	144
FA jacket wall, mm	2.1
Fuel rod per FA, #	126
Fuel rod pitch (triangular), mm	12.2
Fuel rod overall length, mm	2572 (2553.5**)
Fuel stack length, mm	2410 ÷ 2430 (2420)*
Fuel weight, g	1065 ÷ 1109 (1087)
Rod Free Volume (average), cm^3	16.7 ÷ 17.2
Plenum split bush (fixing the fuel stack), #	2
Bush height, mm	10
Fill gas composition	He (98%)
Backfill gas pressure, MPa	0.6 ± 0.1
Fuel Rod Cladding	
Material type	Zr-1%Nb alloy (E110)
Tube outer diameter, mm	9.05 ÷ 9.20 (9.12)
Tube inner diameter, mm	7.72 ÷ 7.80 (7.755)
Wall roughness, µm	~1.5
Fuel Pellets	
Material type	UO_2
Enrichment, wt.% ^{235}U	4.4
Pellet type	Annular
Pellet height, mm	8 ÷ 14 (~11.2)
Pellet outer diameter, mm	7.30 ÷ 7.60 (7.56)
Pellet inner diameter, mm	1.20 ÷ 2.00 (1.65)
Pellet outer diameter roughness, µm	2.0 ÷ 3.0
Pellet grain size, µm	~6.0 ÷ 8.0 (~7)
Fuel open porosity, %	~1.0 ÷ 1.5 (1.25)
Fuel density, g/cm^3	10.4 ÷ 10.7 (10.6)
Fuel densification during thermal testing, vol. %	0 ÷ 1.2 (can be up to 1.8)

* The average (best estimate, BE) values are presented in the parenthesis.

** There is the average rod length w/o the length of the bottom end plug tail.

- the cladding oxide thickness. As noted in report [6, 7], the oxide film thickness on the external surface of the rods examined was in the range of (2 ÷ 7) μm with the maximum located in the upper part of fuel stack. The oxide thickness measurements of rod inner surface showed that oxide film reaches (8 ÷ 17) μm;
- the fuel geometry. There are no systematical experimental data of fuel volume change ($\Delta V/V_0$) vs. rod burnup. However, as mentioned in the accompanying documents for WWER-440 fuel rods [9], the data of $\Delta V/V_0$ vs. fuel burnup are the same as for WWER-1000 fuel and show an increase in $\Delta V/V_0$ to 2.9 % and 3.8 % at the fuel burnup of ~50 GWD/MTU and ~60 GWD/MTU, respectively. The measurement uncertainty is not reported.

1.2.2 Transient Conditions Simulated in Reactor MIR

After the base irradiation, three FRs from FA-198 and six FRs from FA-222 were refabricated and used in the experiments simulating transient conditions. The main characteristics of the refabricated fuel rods (RFRs) are listed in Table 1-2. Some of the RFRs were instrumented by wolfram-rhenium thermocouples by ∅1.2 mm located in the pellet central hole and some of them by pressure transducers (see Table 1-2).

The experiments were carried in the research reactor MIR under single ramp conditions (test referred as “RAMP”) and step-by-step power increase (tests referred as “FGR-1” and “FGR-2”). Variations of the peak linear heat rate for each RFR during the test are shown in Figure 1-1 through Figure 1-3. The uncertainty of the maximum LHR detected was in the range of (6 ÷ 8) %.

During the “RAMP” test the cooling pressure was held at the level of 12.5 MPa. During the “FGR-1”/“FGR-2” test the cooling pressure grows from 11/9.5 MPa to 13/12.5 MPa, when the rod power increases to the maximum value at the last stage (see Figure 1-2 and Figure 1-3). The error in determination of the cladding temperature was ± 6 ^{0}C.

During the “FGR-2” test the thermocouples of the RFR №50 and №51 failed before the end of the experiment. The thermocouple of Rod 50 broke down during the first stage after 78 hours in operation while the thermocouple of Rod 51 failed after 189 hours just before the upper level of the second stage (see Figure 1-3).

The measured uncertainty of fuel temperature was 2%. The measured uncertainty of rod gas pressure was ±0.4 MPa.

Upon completion of all the tests, the measurements of pellet-to-cladding gap, DCO change and FGR were performed:

.- the FGR was measured for seven RFRs only. For Rods 50 and 51 the measurements were not carried out due to damage of the fuel rods during test rig disassembling. The measured uncertainty of FGR is not presented in the final test report;

- the data of the maximum value of pellet-to-cladding gap vs. peak LHR for the WWER-440 rods are available in the final test report. The measured uncertainty was ± 10 μm. The same uncertainty is for the measurement of the cladding outer diameter.

Table 1-2. Characteristics of refabricated WWER-440 fuel rods

"RAMP" Test

RFR number	**33**	**37**	**38**
Full scale FR number/FA number	76/198	5/222	3/222
Burnup, GWD/MTU	50.8	60.1	60.2
Fuel stack length, mm	950	950	950
Outer cladding diameter, mm	9.05	9.04	9.07
Pellet inner diameter, mm	1.6	1.6	1.6
Pellet- to-clladding gap (radial), µm	4-15	0-7	0-7
Filling gas	He	He	He
Backfill gas pressure, MPa	1.1	1.1	1.1
Total free volume, cm^3	6.53	6.03	5.82
Equipping	No	No	No

"FGR-1" Test

RFR number	**41**	**32**	**48**
Full scale FR number/FA number	99/198	6/222	2/222
Burnup, GWD/MTU	48.9	60.2	60.5
Fuel stack length, mm	750	950	750
Outer cladding diameter, mm	9.07	9.06	9.06
Pellet inner diameter, mm	1.6	1.6	1.6
Pellet- to-clladding gap (radial), µm	4-15	0-7	0-7
Filling gas	He	He	He
Backfill gas pressure, MPa	2.0	1.1	1.1
Total free volume, cm^3	5.84	6.04	6.07
Equipping	PT*	No	PT*

* PT is the pressure transducer.

"FGR-2" Test

RFR number	**51**	**50**	**52**
Full scale FR number/FA number	20/198	25/222	46/222
Burnup, GWD/MTU	49.5	58.4	58.0
Fuel stack length, mm	400	400	400
Outer cladding diameter, mm	9.035	9.059	9.070
Pellet inner diameter, mm	1.6	1.6	1.6
Pellet- to-clladding gap (radial), µm	4-15	0-7	0-7
Filling gas	He	He	He
Backfill gas pressure, MPa	1.2	1.2	1.2
Total free volume, cm^3	10.2	10.4	9.0
Equipping	TC*	TC*	No

* Thermocouples are located 100 mm below the top of the fuel stack

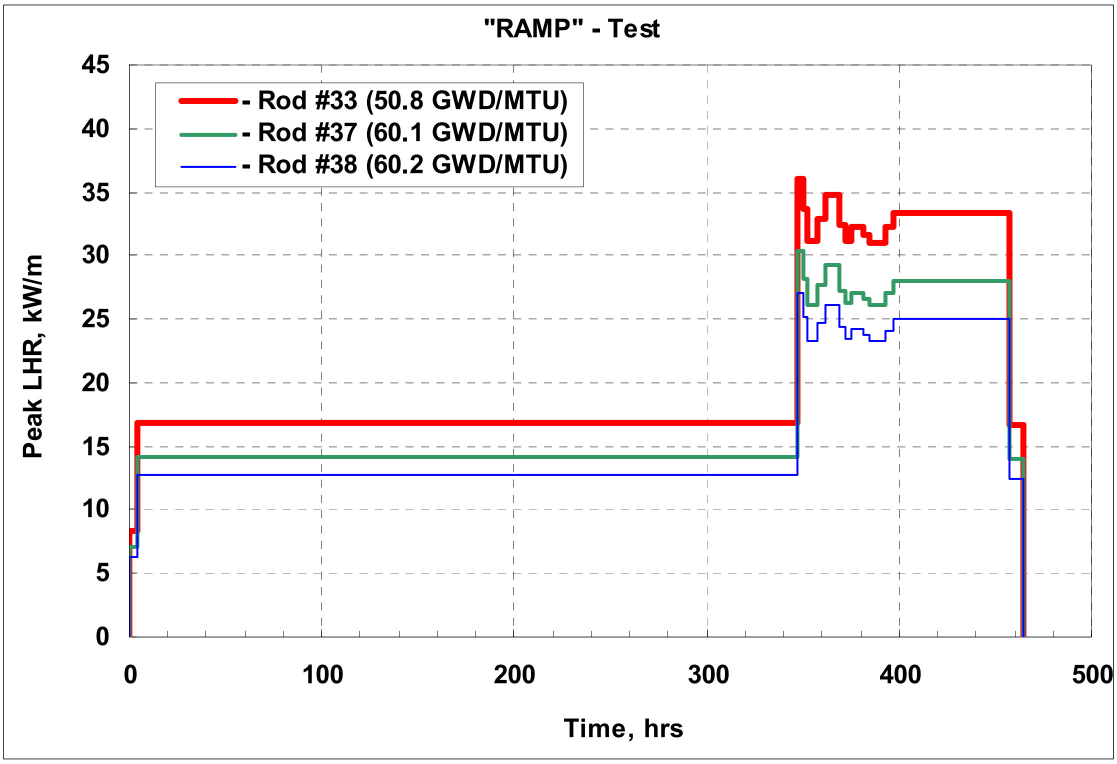


Figure 1-1. Variation of peak linear heat rate for refabricated WWER-440 fuel rods during power ramp test "RAMP" carried out in reactor MIR.

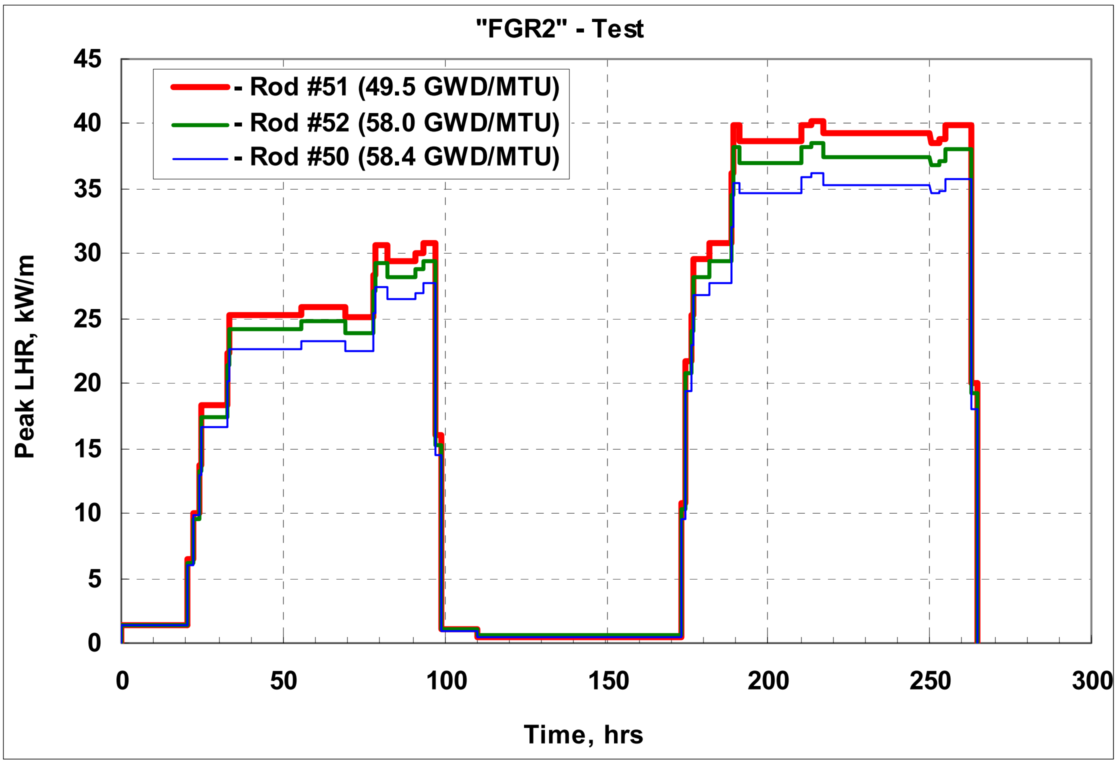
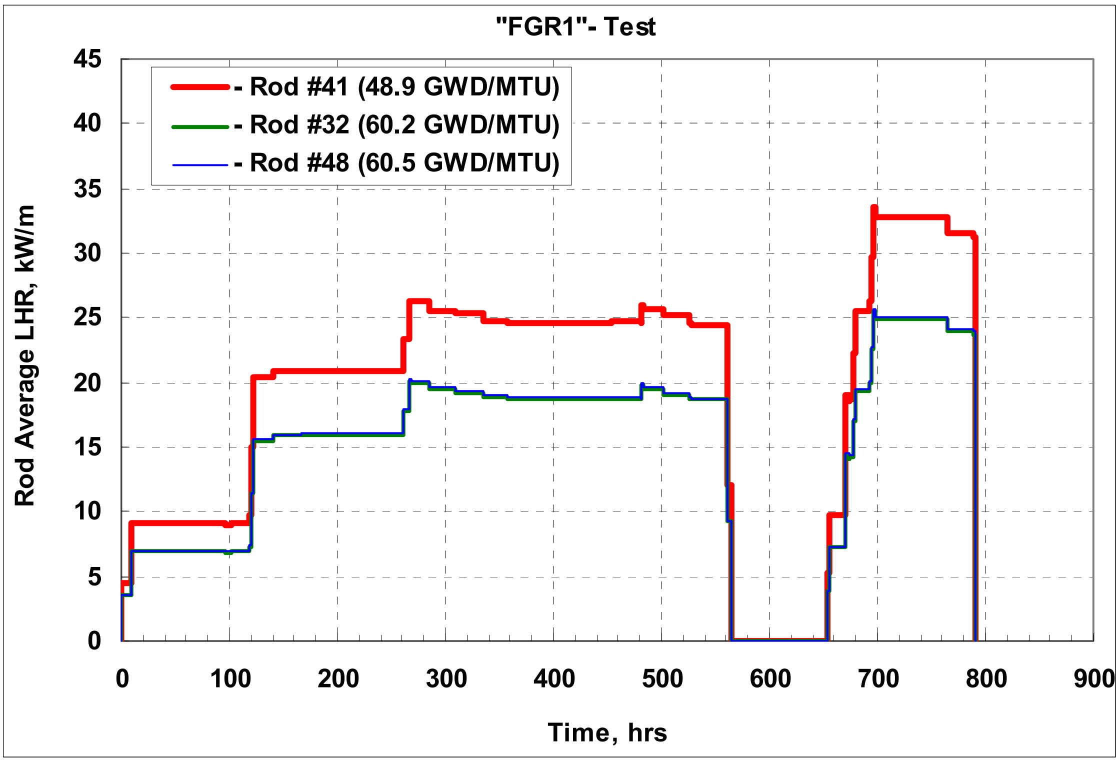


Figure 1-2. Variation of peak linear heat rate for refabricated WWER-440 fuel rods during power ramp test "FGR-2" carried out in reactor MIR.



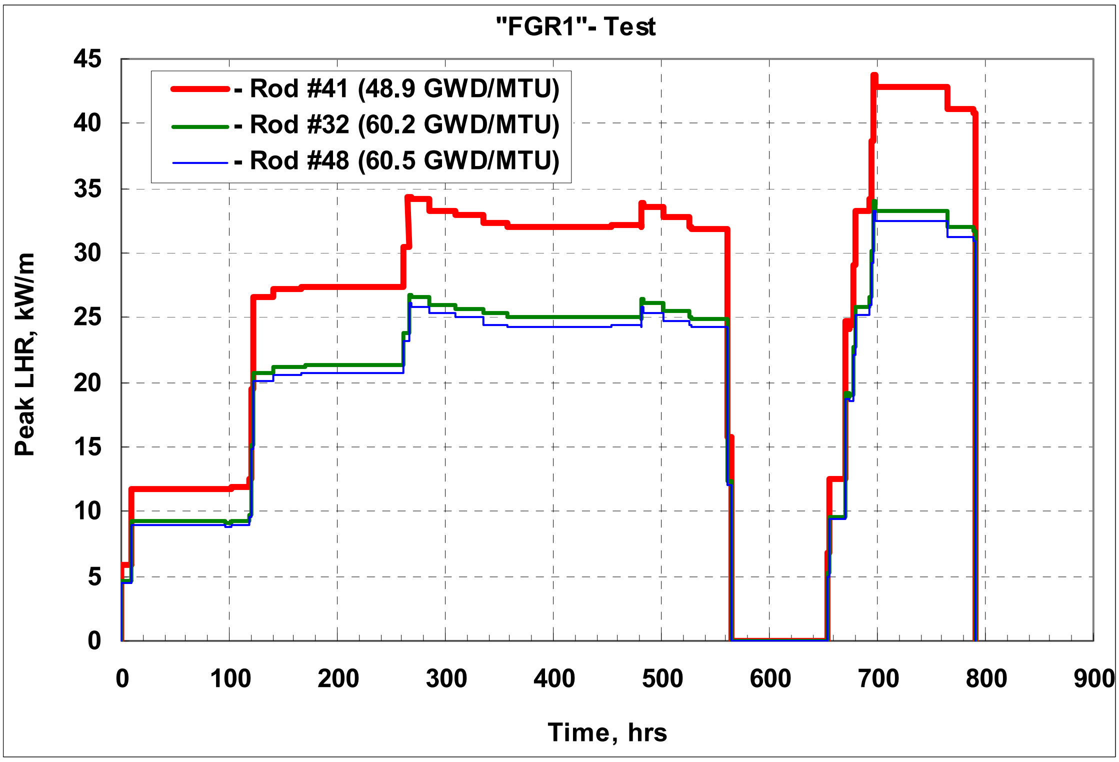


Figure 1-3. Variation of rod average and peak linear heat rates for refabricated WWER-440 fuel rods during power ramp test "FGR-1" carried out in reactor MIR.

1.3 Description of PAD and TRANSURANUS Fuel Rod Models

➢ *PAD code*

The PAD-4.0 version 10.5.2 fuel rod model for a low-tin zirconium alloy cladding applicable to the VVER-1000 fuel design was used as a basis during the current FR analysis.

The description of PAD code models, such as the thermal, corrosion, cladding creep and growth, fission gas release and rod internal pressure, was previously presented in [10]. Below is a description of some model peculiarities related to the WWFR-440 FR design and the operation conditions.

The corrosion model for the low-tin Zr-1%Nb alloy with a modified corrosion rate constant [11, 12] was utilized.

The in-reactor cladding creep model for the low-tin Zr-1%Nb alloy with the increased parameter for the cladding irradiation creep [12] was used to estimate the cladding deformation of the tested FA-198 & 222 FRs.

The code BE fast neutron conversion parameters developed for WWER-1000 core were utilized as the basic ones to calculate the neutron fluence used in the FR irradiation growth, $\Delta L/L_0$. The cladding growth model for the low-tin Zr-1%Nb alloy with the BE model parameters, which showed a good agreement with the experimental data for the WWER-1000 rods with E-110 cladding [12], was utilized in the current FR analysis.

Two fuel swelling and densification models are available in the PAD code. One of them, referred to as "*WEC*" model, was developed for the fuel manufactured using the ADU technology with a sintering temperature of up to 1780^0C. The other one, referred to as the "*NRC*" model, is used when the pellet sintering process differs from the ADU technology. A resinter density change is used as an input parameter in this model. Both models were used in the PAD FR analyses for the WWER-440 FRs.

The code BE parameters for the FGR model were used as the basis, when the normal operation and transient conditions are simulated. Also, the modified gas release model parameters [12] - threshold burnup and temperature for the high-burnup region, were utilized during the calculation of gas release at the transient.

The thermal conductivity (λ) correlation for UO_2, which takes into account the effect degradation during the fuel burnup growth, was used. The sensitivity analysis of thermal conductivity on FCT and FGR was carried out. For this case, the model parameter at the burnup component (λ referred to as *L25*) was varied in the range of ± 20% (λ referred as *L30*/*L20*, respectively).

After the base irradiation the PAD transient option was activated at the time frames of power ramping.

➢ *TRANSURANUS code*

The code TRANSURANUS v1m1j09 (TU) [13], which takes into account the relevant phenomena occurring in the WWER fuel (code parameter "*WER*") and the specific features of the cladding made from E-110 alloy (code parameter "*ZR1*"), was used in the analysis.

The calculations have been carried out coherently with the power histories of WWER-440 rods examined and using the BE fuel rod parameters from Table 1-1 and the some characteristics of RFRs from Table 1-2. The value of the fuel pore removable during sintering of 1.2 % (code parameter "*resint*") and burnup at which sintering has stopped of 5 GWD/MTU, based on the ITU investigations [14, 15], were used.

Below, the following TU-WWER fuel rod models were used -

- the simple empirical model of fuel densification and swelling [13];
- the formation of fuel cracks, which are considered as an adder to the rod free volume. The BE parameter of fuel crack model is applied;
- the fuel pellet-to-cladding interaction with the BE static and sliding friction coefficients and without accounting for slip for axial PCMI;
- the grain growth (model of Ainscough and Olsen);
- the fuel relocation model. The modified KWU-LWR model (code parameter *Ireloc*= 5) and the modified FRAPCON-3 model (code parameter *Ireloc*= 8) were utilized based on the results presented in [13];
- the intragranular fission gas release (*URGAS* model with the BE thermal and athermal diffusion coefficients, proposed by Hj. Matzke);
- the intragranular fission gas release with the grain boundary gas saturation concentration of 1E-4 mol/m^2, as the basic one;
- the FGR from the fuel high-burnup structure (*HBS* model with the BE threshold burnup of 85000 MWD/MTU);
- the cladding material properties, which take into account the specific correlation for Zr-1%Nb alloy [13] (parameter *ModClad*(4)=25). Also, the texture factors for irradiation growth of E-110 cladding were used based on the recommendation [13] and they are f_R = 0.52, f_T = 0.38, f_A = 0.10;
- the UO_2 material properties presented by the standard TU models for LWR fuel (parameter *ModFuel*(4)=20) [13];
- Also, two correlations for thermal conductivity of UO_2 - *Lamf*=21 and *Lamf* =25, were used. The correlation of *Lamf*=21, which is the standard TU correlation for UO_2 and $(U,Gd)O_2$, was used as the basis

to simulate the FR performances at the steady-state operation and the transients conditions.

Using the restart options of TRANSURANUS, the changes in the filling gas and pin pressure after the base irradiation were accounted for.

1.4 Simulation Results of WWER-440 Fuel Rod by PAD and TRANSURANUS

1.4.1 Fuel Burnup

The calculations of the rod average (<BU>) and the maximum (peak) burn-ups for the FA-198 and FA-222 FRs revealed that the predicted burnups obtained by both codes lie within the range of the measured uncertainties (see Figure 1-4).

- *PAD code*

On the whole, a slight underestimation of <BU> and overestimation of BU_{max} is observed. For the nominal characteristics of WWER-440 FR used, the deviation between the measured and calculated <BU> averaged over all rods examined is 0.8 % (S.D. = 1.9 %). For the maximum fuel burnup the deviation is -3.1 % (S.D. = 2.5 %). In the case, when the maximum pellet diameter (DP =7.59 mm) is used, the deviation of <BU> reaches 1.6 % (S.D. = 1.8 %), while the deviation of BU_{max} decreases to -2.2 % (S.D. = 2.5 %).

- *TRANSURANUS code*

There is a tendency of underestimation with the rod burnup growth (see Figure 1-4). For the nominal design characteristics of WWER-440 rod used, the deviation between the measured and the code predicted burnup is - 0.2 % (S.D. = 2.4 %) and 3.6 % (S.D. = 1.8 %) for the <BU> and BU_{max}, respectively.

1.4.2 Fuel Volume Change

There are no data on the fuel volume change ($\Delta V/V_0$) versus burnup, which will be available for the examined FRs of FA-198 & 222 assemblies. However, as mentioned in [6, 16], the fuel swelling of WWER-440 and WWER-1000 FRs is the same. This is supported by some data from the report for WWER-440 RFRs tested which demonstrate that the $\Delta V/V_0$ value at the rod burnup of ~50 GWD/MTU is ~2.9 % and increases to 3.8 %, when the burnup grows to ~60 GWD/MTU [9].

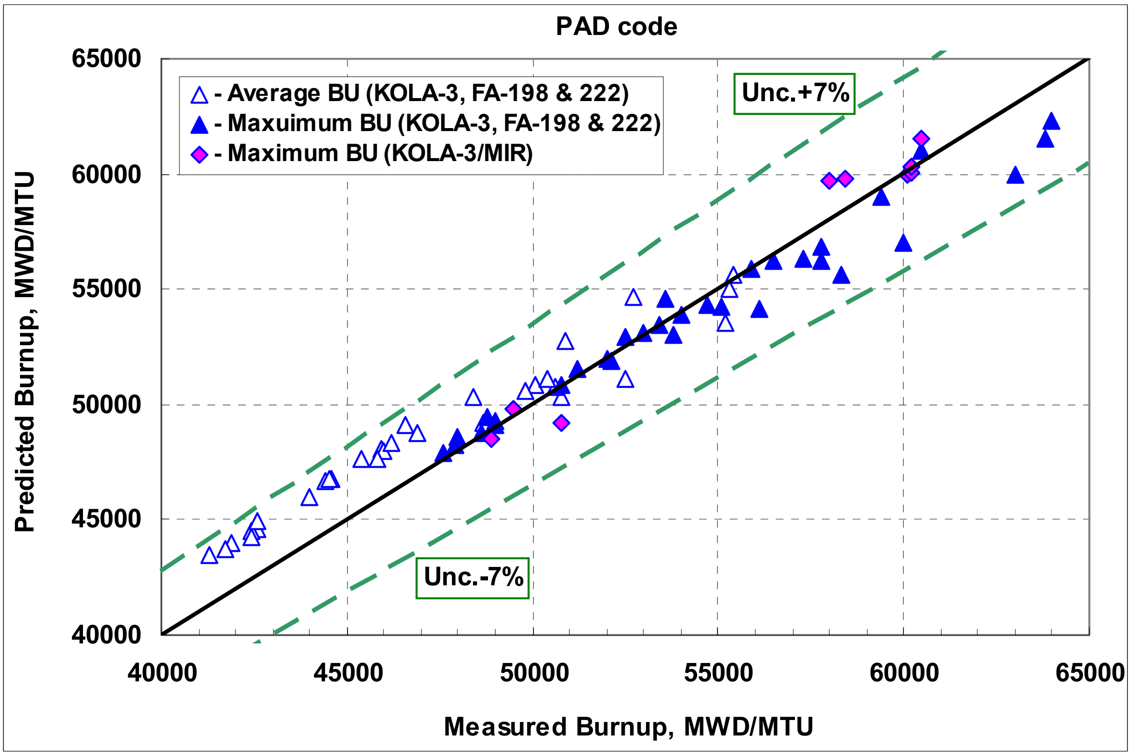
- *PAD code*

To estimate the code prediction capability of fuel swelling for WWER-440 FR, the data of $\Delta V/V_0$ presented above were used. The calculations were carried out using both "WEC" and "NRC" models assuming isotropic fuel densification and swelling. A pellet sintering temperature of 1700 °C [8] was used in the "WEC" model. The resinter density change of 1.2 % was utilized, when the "NRC" model is applied.

The performed FR analysis revealed that for the "WEC" model used the calculated $\Delta V/V_0$ agrees well with the measured ones. The deviation between the measured and the predicted $\Delta V/V_0$ at the fuel burnup of 61 GWD/MTU does not exceed 0.3 %. In the case when the "NRC" model is used, the calculated $\Delta V/V_0$ shows overestimation with burnup growth. The deviation value at the burnup of 61 GWD/MTU is about -1 %. The increase fuel densification rate up to 1.8 % leads to decrease in the deviation to -0.6%.

- *TRANSURANUS code*

- on the hole, satisfactory agreement is observed when the low bound of fuel densification is utilized (parameter "*resint*"=1.2). At the fuel burnup of 61 GWD/MTU, the deviation between the measured and the predicted $\Delta V/V_0$ is about -0.5%.
- in the case, when the fuel densification of 1.8 % is used, ~0.9% underestimation is observed at the burn-ups of 35÷40 GWD/MTU. As burnup grows, the deviation between the measured and the predicted $\Delta V/V_0$ decreases. In the burnup range of (55 ÷ 61) GWD/MTU the predicted $\Delta V/V_0$ values lie within the experimental database.



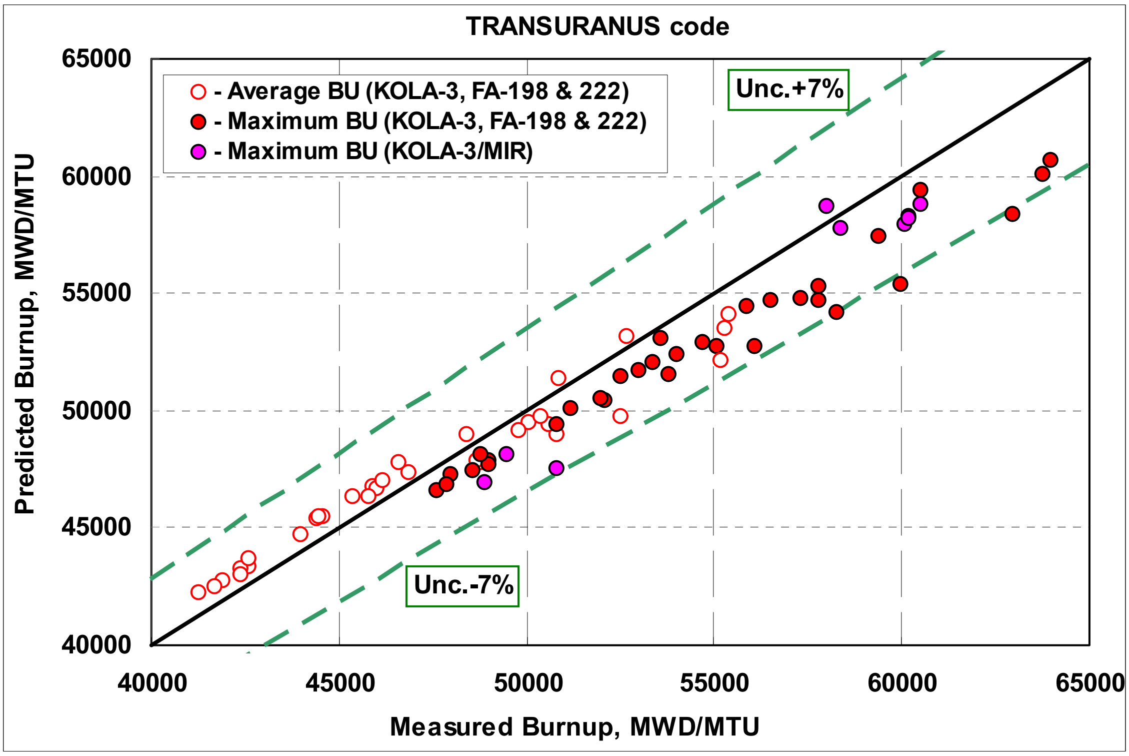


Figure 1-4. Code predicted burn-ups vs. measured burn-ups for FA-198 and FA-222 rods after base irradiation in WWER-440 core of reactor Kola-3.

1.4.3 Cladding Corrosion

The analysis results of oxide thickness variation along the fuel stack calculated by both codes showed that the maximum oxide film is reached in the range of (80-90) % of the fuel stack length. This result agrees with the data mentioned in [6, 7]. The code specific results are listed below.

- *PAD code*

The calculated of the peak oxide thickness for the FA-198 & 222 FRs examined is varied in the range of (4.5 ÷ 6.7) µm and are in satisfactory agreement with the experimental data. The maximum oxide thickness predicted for the FRFs after the base irradiation lie in the range of (2.0 ÷ 3.7) µm.

- *TRANSURANUS code*

The maximum peak oxide thickness predicted for the FA-198 & 222 FRs examined is varied in the range of (7.0÷9.0) µm. This results lie within the experimental data for WWER FRs [6, 7].

1.4.4 Fuel Central Temperature

- Steady-State Operation

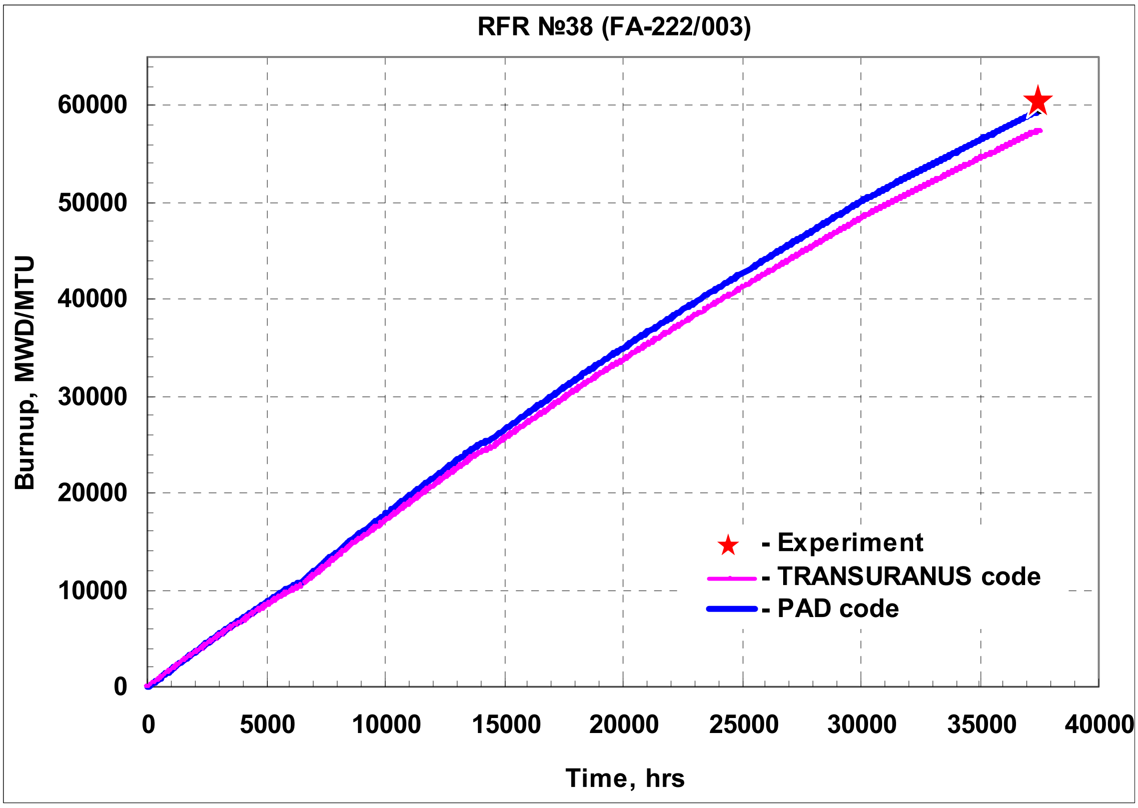
Obviously, no fuel temperature measurements could be made for the considered WWER-440 FRs during core operation. Based on the PAD and TRANSURANUS analyses of the FA-198 & 222 FRs, the maximum fuel centerline temperature (FCT) is reached during the 1st cycle and at BOC-2 and the maximum FCT stays below 1000 ^{0}C. To compare the maximum FCTs calculated by PAD and TU codes, the additional analysis was performed. Calculations were carried out for RFR №38 having the maximum burnup at EOL. Dependence of the maximum BU vs. time operation calculated by both code is shown in Figure 1-5. The time variation of the maximum FCTs for the different λ(T) correlations for UO_2 are presented in the same figure. The obtained results demonstrate:

- for a ”fresh” fuel, at the peak LHR of 200 W/cm the FCT does not exceed 880 ^{0}C. The TU correlations, *Lamf*=21 and 25, provide the higher FCT in comparison with the PAD code. The difference is ~100 ^{0}C and decreases to 85 ^{0}C at the fuel BU of ~12 GWD/MTU;
- starting from BU ≈26 GWD/MTU, the difference between FCTs calculated by the TU code with λ(*Lamf*=21) and the PAD code with λ(L25) is about 60 ^{0}C and decreases to (30÷40) ^{0}C in the high burnup range. In the case, when the PAD code parameter L30 is utilized, the difference of FCT shows a two-fold decrease at BU ≈ 60 GWD/MTU;
- in the burnup range of (50 ÷ 60) GWD/MTU, the FCT calculated by two codes varies in the range of (575 ÷ 640) ^{0}C at the LHR of ~120 W/cm. The TU correlation of *Lamf*=25 shows the lowest FCT values.

- Ramp test

The PAD and TU analysis of FCT behaviour during the power ramp test (“FGR-2”) revealed:

- the calculated FCTs are in satisfactory agreement with measured ones only for the RFR №51, which started the irradiation with lower initial burnup, 49.5 GWD/MTU (see Figure 1-6). The performed PAD sensitivity analysis showed that λ(T)-correlation with the model parameter of *L20* already provides a good agreement of the calculated FCTs with the measured ones. The same is observed for TU *Lamf*=25 correlation;
- for the RFR №50, the FCT values recorded before the thermocouple failure are noticeably higher than the ones calculated by both codes (see Figure 1-7). The deviation of FCT stays in the range of (190 ÷ 300) ^{0}C, when the rod power varies from 21.3 kW/m to 25.1 kW/m.



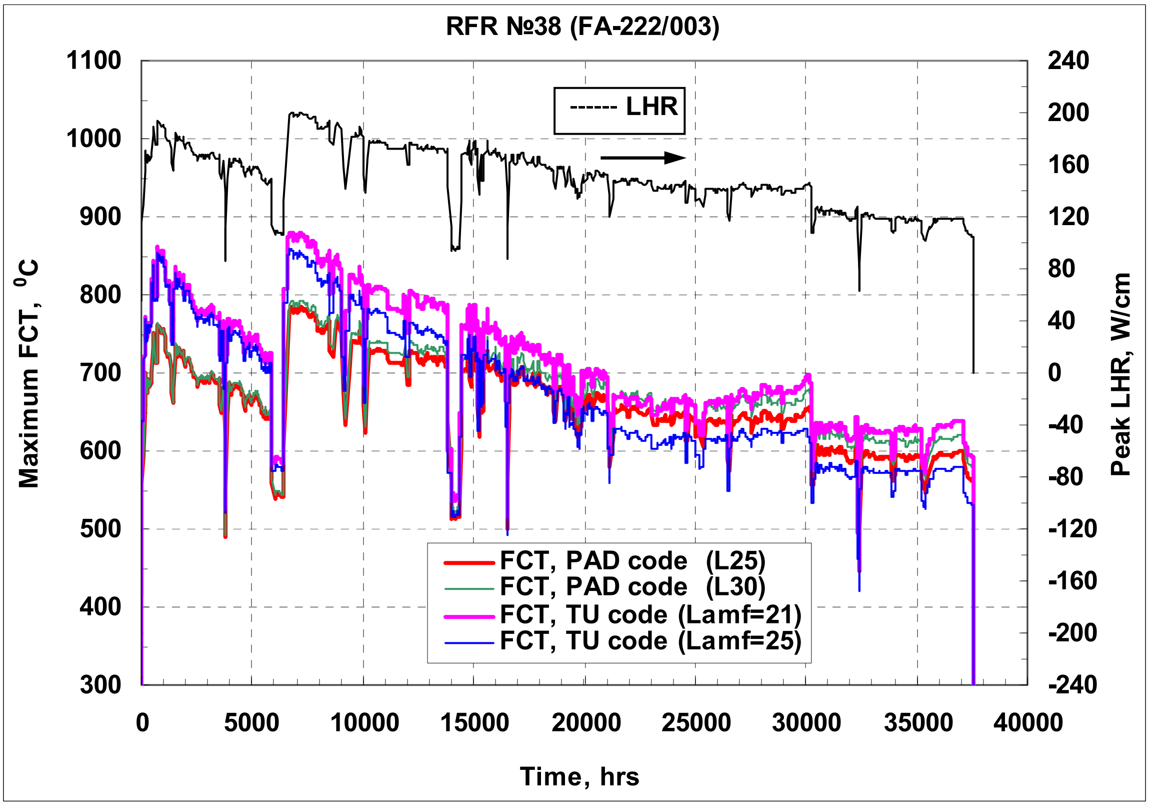
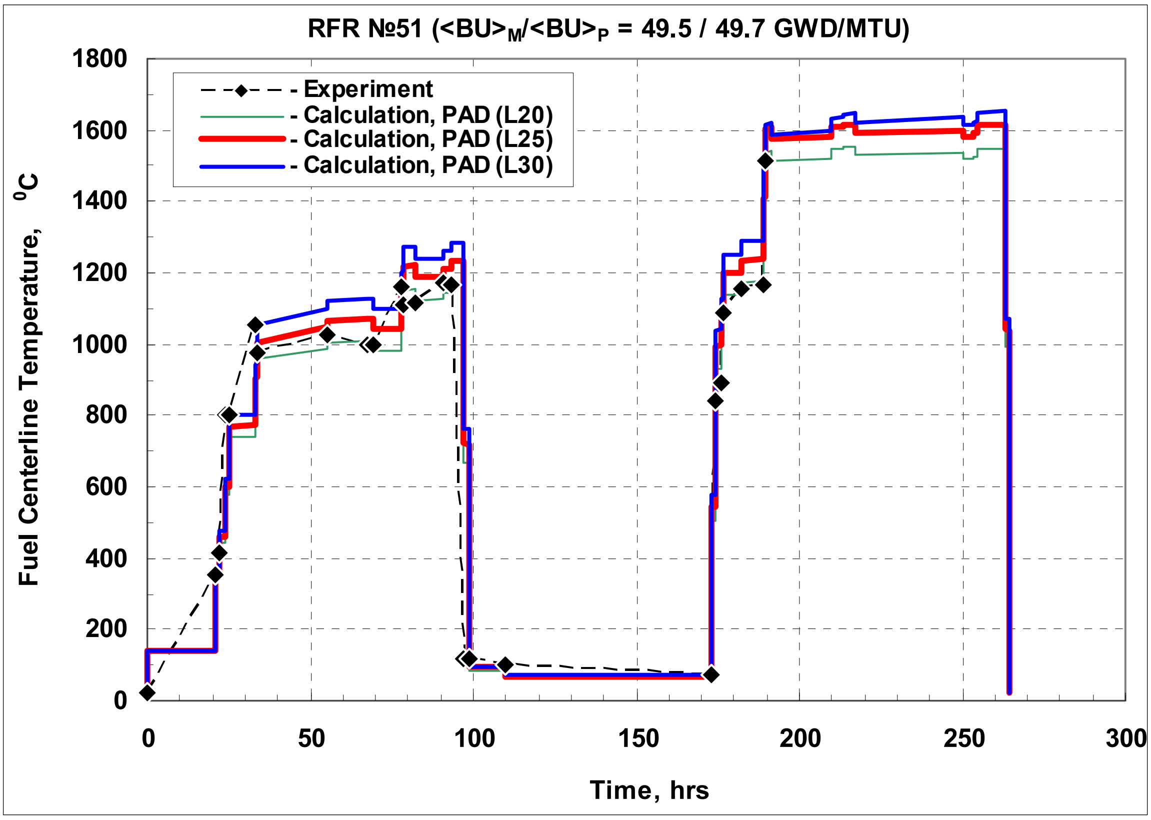


Figure 1-5. PAD and TRANSURANUS calculation results of maximum fuel burnup and maximum fuel centerline temperature for refabricated fuel rod №38 (FA-222/003) during steady-state operation in reactor Kola-3. The FCT is calculated for peak LHR.



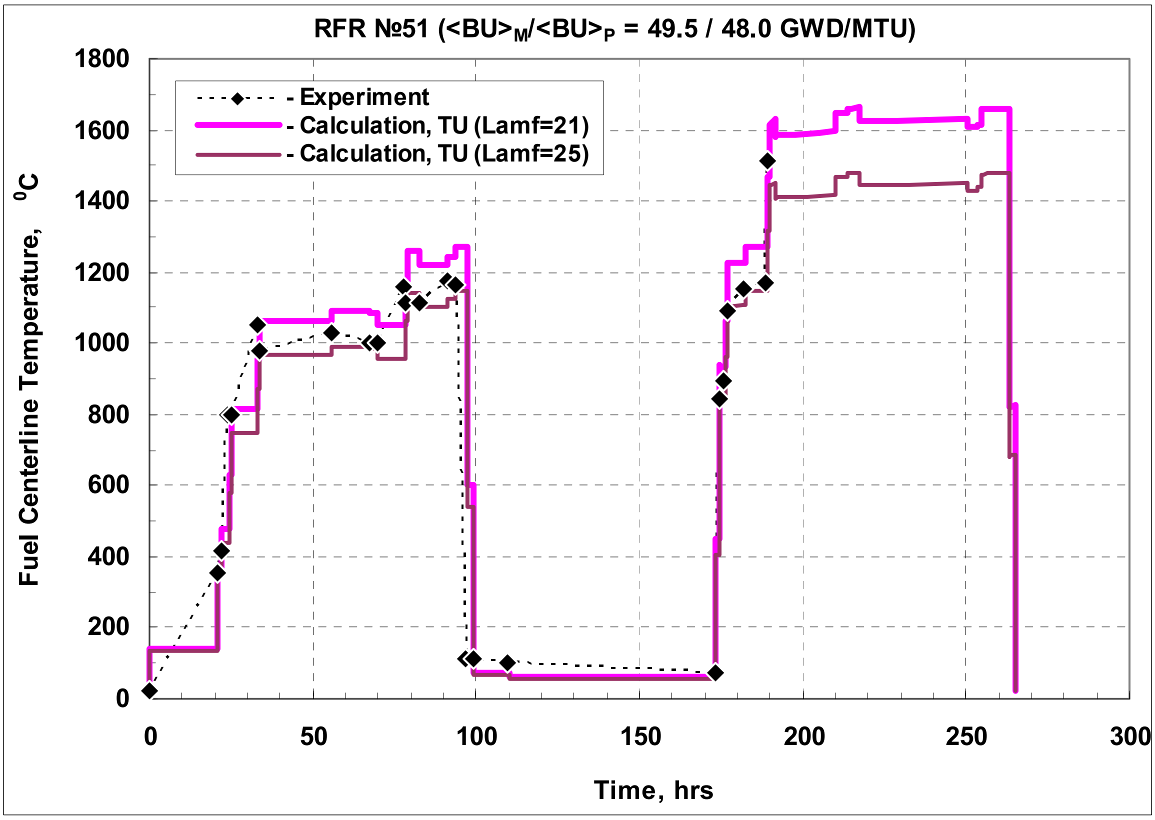
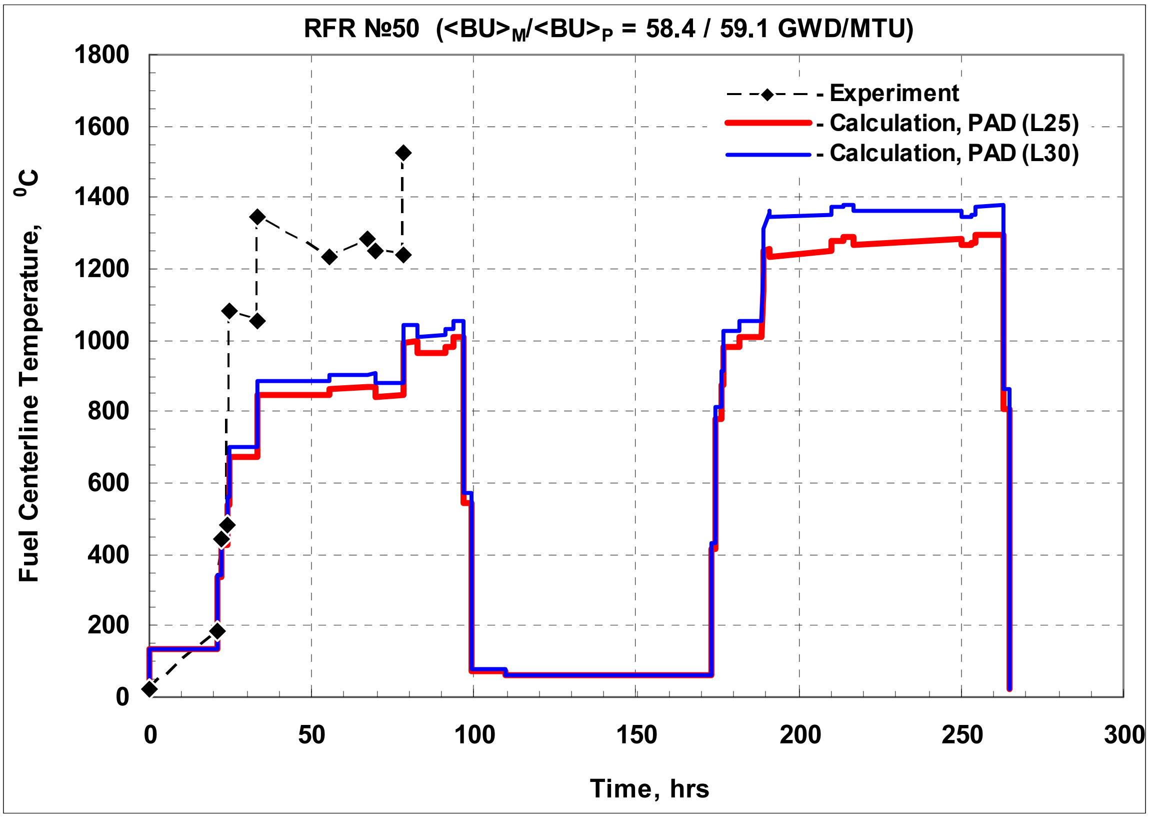


Figure 1-6. Measured and code-predicted fuel centerline temperature variation for refabricated fuel rod №51 during power ramp test "FGR-2" carried out in reactor MIR

(<BU>$_M$ and <BU>$_P$ is the measured and the code predicted rod average burnup, respectively)



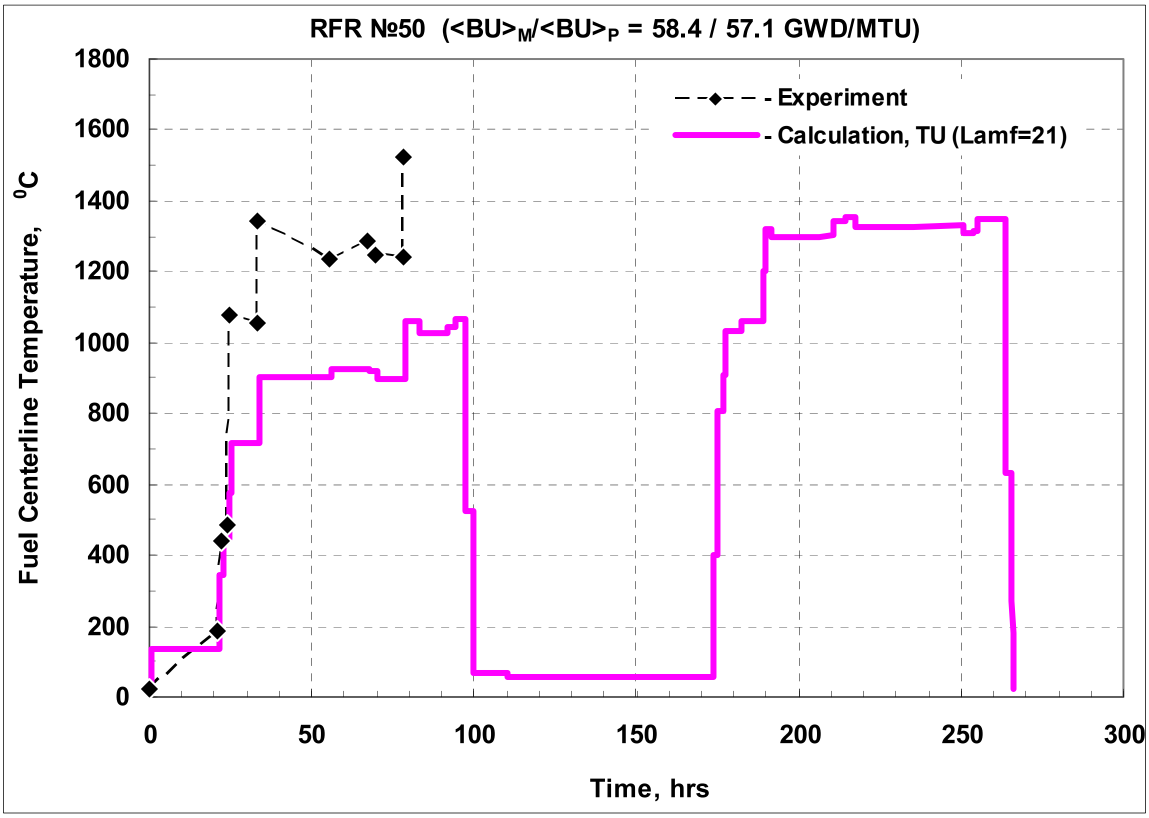


Figure 1-7. Measured and code-predicted fuel centerline temperature variation for refabricated fuel rod №50 during power ramp test "FGR-2" carried out in reactor MIR

($\langle BU\rangle_M$ and $\langle BU\rangle_P$ is the measured and the code predicted rod average burnup, respectively)

1.4.5 Cladding Outer Diameter Change

It is well known that the cladding creep and fuel swelling determine the value of cladding outer diameter change (ΔDCO) and pellet-to-cladding diametral gap during the irradiation. At the BOL operation, when there is a gap between the cladding and fuel pellets, the cladding creep-down due to pressure drop leads to decrease in DCO. Under irradiation, both solid and gas swelling of fuel result in increase in the pellet outer diameter. At some burnup, the diametral gap at power begins to be closed due to fuel swelling and thermal expansion. This pellet-cladding mechanical interaction (PCMI), referred to as "soft" contact, does not impact pellet cracks, which were formed at the early stages of operation. As fuel burnup grows, the fuel cracks start to close backwards due to increasing pellet surface contact load and, at some threshold burnup (BU_0), the fuel pellets may be considered as intact. As a result, the "hard" contact between the fuel and the cladding is realized. Starting from the BU_0, the cladding is subjected to tensile load due to fuel swelling and the value of DCO begins to increase.

In the case, when the power transient occurs, the thermal expansion of fuel pellet volume due to a sharp increase in FCT will increase the cladding outer diameter. The residual value of cladding strain (ΔDCO) as well as the diametral gap after the transient depends on several factors. The main of them are the fuel burnup and the initial diametral gap before the transient, cladding thickness, transient duration and ramp power level. Certainly, the burnup-depending cladding strength characteristics and the cladding creep-out value will affect the gap and the cladding outer diameter change.

- <u>Steady-State Operation</u>

The profilograms of cladding outer diameters measured for the rods #096 (FA-198) and #001 (FA-222) after the base irradiation to the average burnup of 41.7 and 55.4 GWD/MTU are shown in Figure 1-8. For these rods the decrease in the cladding outer diameter in the FR active part is reported as ~70 μm and ~40 μm, respectively. The calculated data of the axial variation of the cladding outer diameter obtained by means PAD code are shown in the same figure. During the calculation the initial DCO of 9.11 mm for the rod #096 and 9.15 mm for the rod #001 were used (see Figure 1-8). The data presented in the figure allow making the following conclusions:

- the corrosion and the in-reactor cladding creep models used in the current analysis provide an acceptable agreement between the calculated and measured DCO;
- at the high burnup (55.4 GWD/MTU), when the "hard" contact between fuel and cladding is registered [6], the PAD FR models properly simulate the DCO change due to PCMI inducing the cladding tensile strain;
- at the given fuel density of ~96.7 %T.D., the variation of initial rod component dimensions within the manufactured uncertainties and the pellet densification rate can affect the DCO change. The PAD sensitivity analysis performed for Rod #001 showed that there is satisfactory agreement between the measured DCO and the calculated one, when the average pellet diameter of 7.56 mm and the "NRC" fuel swelling models are used. The same result is also observed, when DP= 7.59 mm together with "WEC" model are utilized.

The dependence of the averaged value of DCO change measured in the fuel rod active part height versus the maximum fuel burnup is shown in Figure 1-8. The codes calculation results are presented in the same figure. As a whole, the measured and the predicted values are in good agreement, accounting for the measurement uncertainty of ±10 μm.

The TRANSURANUS results give a smaller diametrical change (-38 ÷-44) μm, which does not significantly vary, when different fuel relocation (code parameter *Ireloc*=5/8) and cladding creep (*ModClad*(4)=25 or 27) models are used. As can be seen, TU results show better agreement with the experimental data in the fuel burnup range of (55 ÷ 65) GWD/MTU.

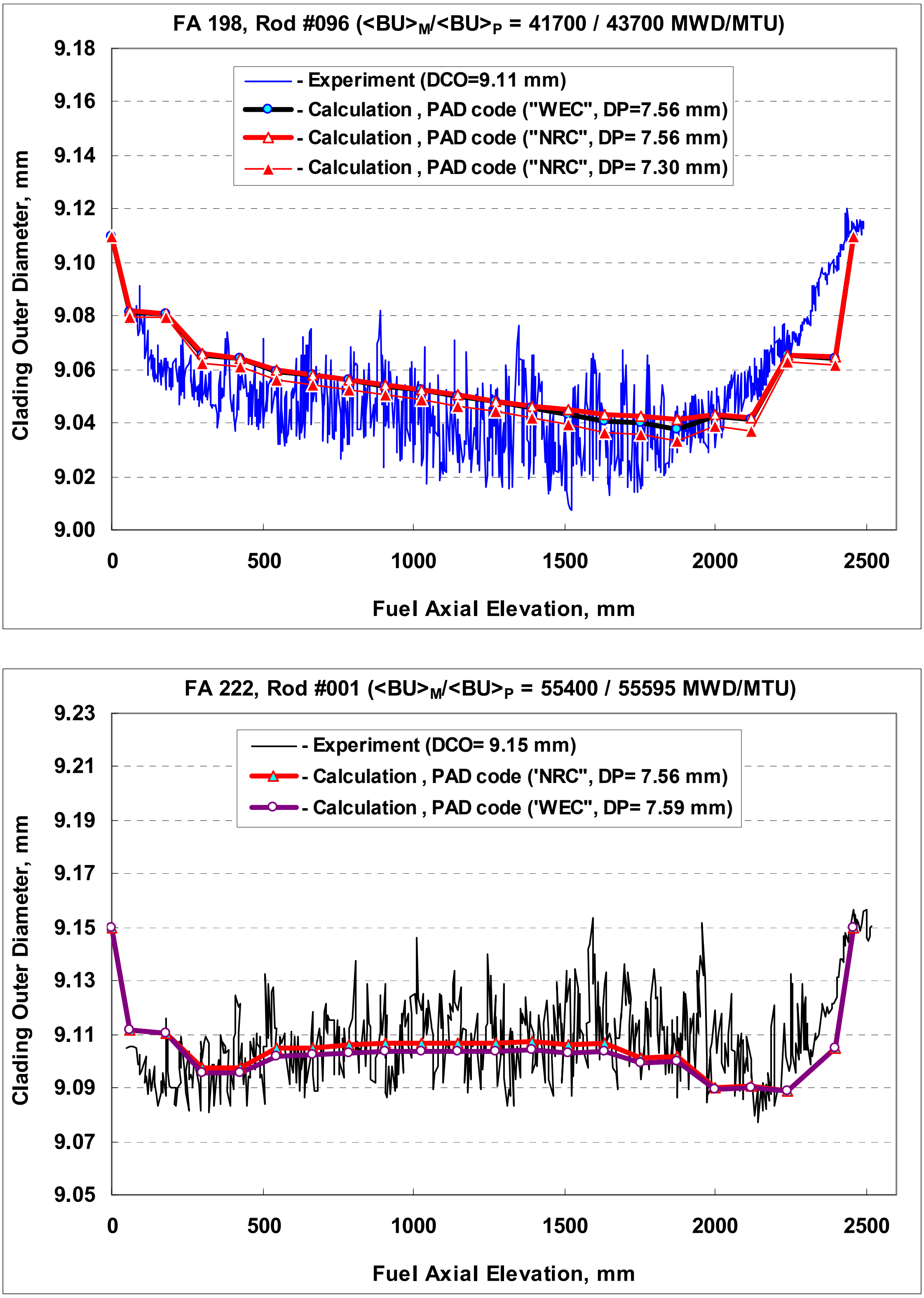


Figure 1-8. Measured and PAD predicted axial variation of cladding outer diameter for FA-198 & 222 rods after base irradiation in reactor Kola-3

(<BU>$_M$ and <BU>$_P$ is the measured and the code predicted rod average burnup, respectively)

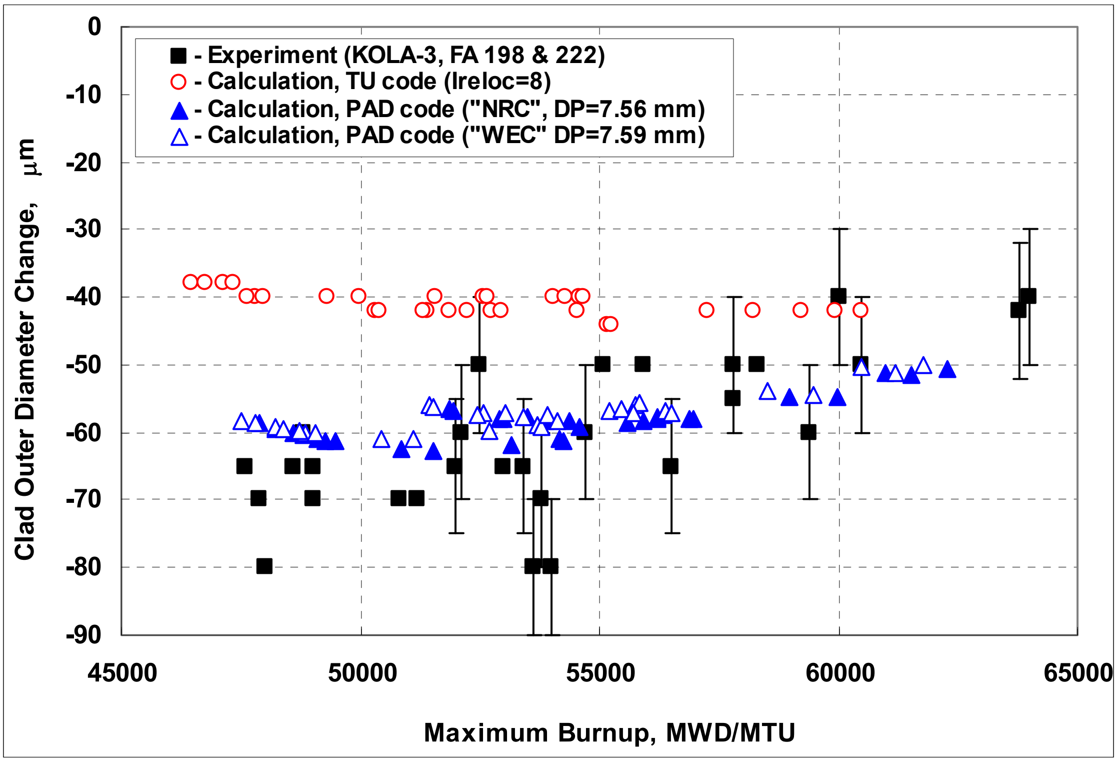


Figure 1-9. Dependence of cladding outer diameter change versus maximum fuel burnup measured for FA-198 & 222 rods after base irradiation in reactor Kola-3 and calculated by TRANSURANUS (TU) and PAD codes

The PAD results give a more reasonable compliance with the measurement. As already mentioned, the calculated value of the maximum cladding outer diameter decrease (ΔDCO) obtained by using the "WEC" fuel swelling model with the initial pellet diameter of 7.59 mm are close to the one obtained by "NRC" model and this is observed in Figure 1-9.

For the FA-198 & 222 FRs examined, the deviation between the measured ΔDCO and the PAD calculated data averaged over all fuel rods is -2.5 µm and -3.5 µm (S.D. ≈ 10 µm) for the "NRC" and the "WEC" fuel swelling model, respectively;

- <u>Ramp test</u>

The measurement results of cladding outer diameter change (ΔDCO) versus the maximum (peak) LHR obtained for the refabricated WWER-440 rods after the power ramp tests are shown in Figure 1-10. The data of ΔDCO calculated by the PAD and the TRANSURANUS codes are presented in the same figure. As can be seen, the measured and the predicted values are in satisfactory agreement and they evidence:

- both codes predict the cladding outer diameter increase with LHR growth;
- the TRANSURANUS calculation results drop within the measured uncertainties, when the LHR exceeds 375 W/cm;
- PAD results show ~20 µm overestimation, when the maximum LHR of 450 W/cm is reached.

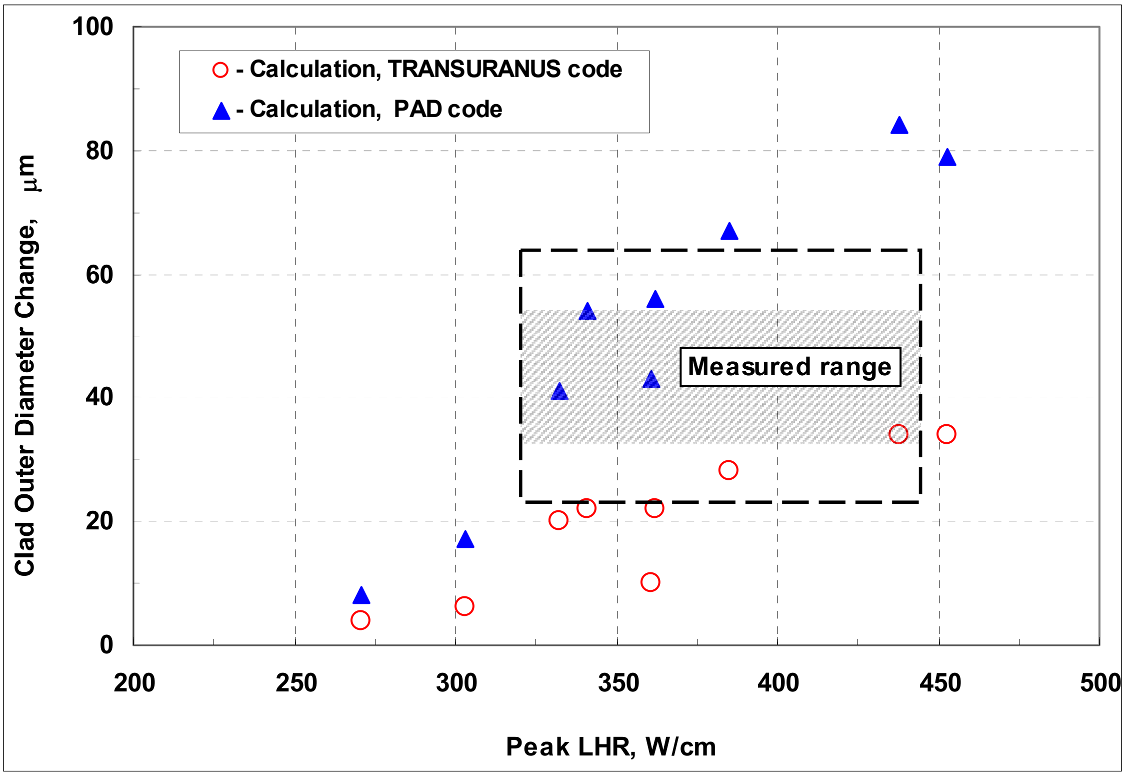


Figure 1-10. Dependence of cladding outer diameter change after power ramp versus maximum LHR measured for refabricated rods (FA-198 & 222) irradiated in reactor MIR and calculated by PAD and TRANSURANUS codes

1.4.6 Fuel-to-Cladding Gap at Fuel Active Height

- Steady-State Operation

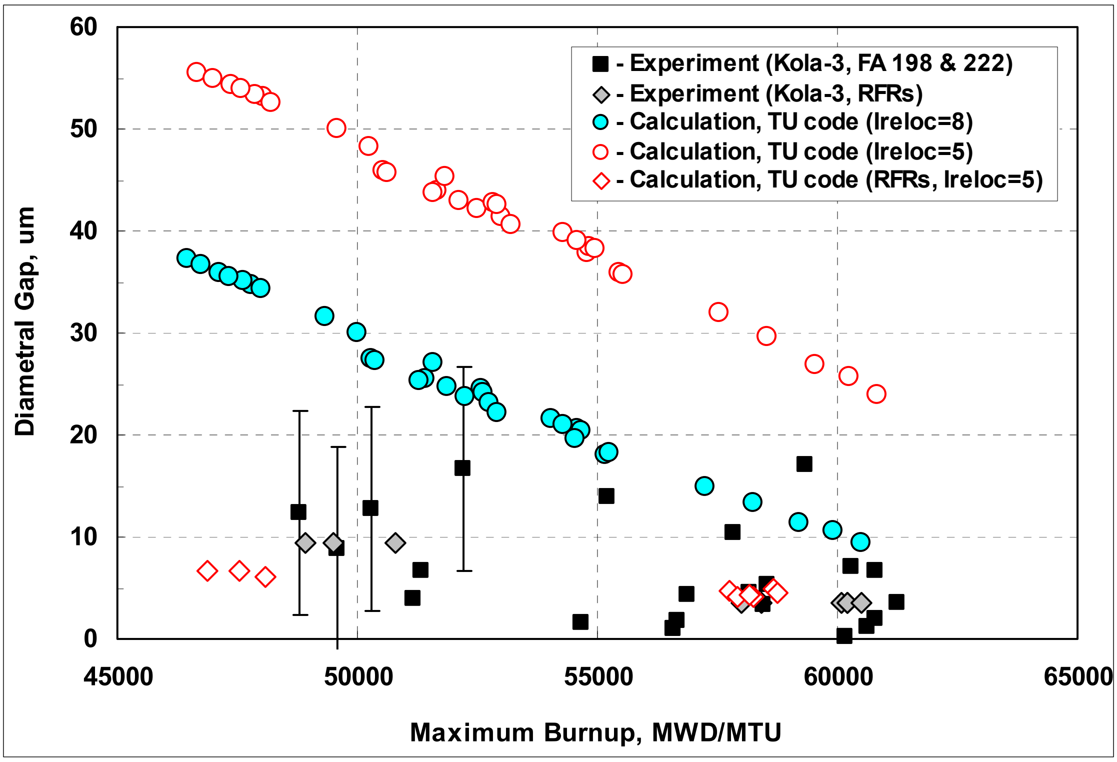
Comparison of the calculated and the measured diametral gap for the FA-198 & 222 FRs examined after the base irradiation are shown in Figure 1-11. Accounting for the measurements uncertainty of ±10µm and for the big manufacturing tolerances of the cladding inner diameter (+45/-35 µm) and pellet diameter (DP, +40/-30 µm), it may be concluded that the predicted values are in satisfactory agreement with the experimental data.

➢ *TRANSURANUS code*

The calculation results obtained by using different fuel and WWER cladding models showed that the FRAPCON-3 fuel relocation model (*Ireloc*=8) provides better agreement with the experimental data for the full-scale FA-198 & 222 FRs. In the case, when the RFRs are simulated, the good agreement with experimental data are also obtained when the KWU-LWR relocation model (*Ireloc*=5) is utilized (see Figure 1-11). The same result was found in [12].

➢ *PAD code*

The PAD calculation results give a more reasonable compliance with the measurement. For both fuel swelling models utilized ("WEC" & "NRC") the calculated values lie within the measured uncertainties. In the burnup range of (55 ÷ 62.5) GWD/MTU the calculated gap values at the normal condition for the full scale and the refabricated FRs vary from 2 µm to 12 µm (sse Figure 1-11). This result correlates with the experimental database and evidences that the "hard" contact between the pellets and the cladding is realized along the fuel rod active part height.



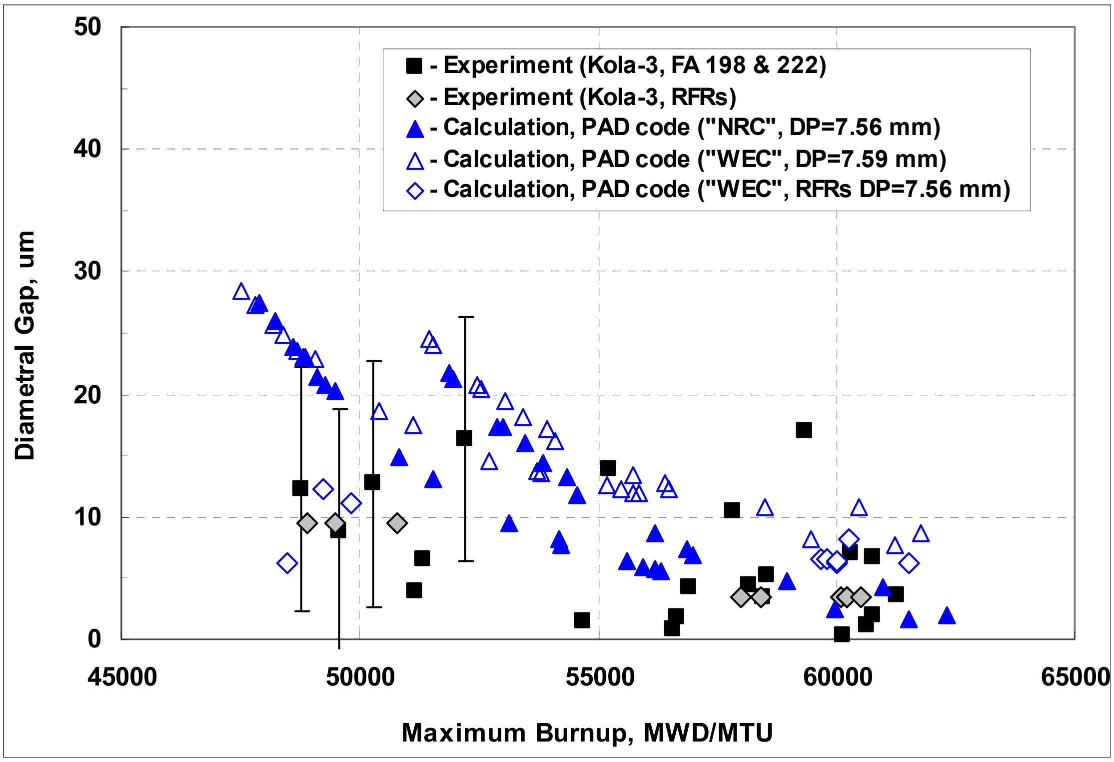


Figure 1-11. Dependence of pellet-to-cladding diametral gap change versus maximum fuel burnup measured for WWER-440 rods after base irradiation in reactor Kola-3 and calculated by TRANSURANUS and PAD codes.

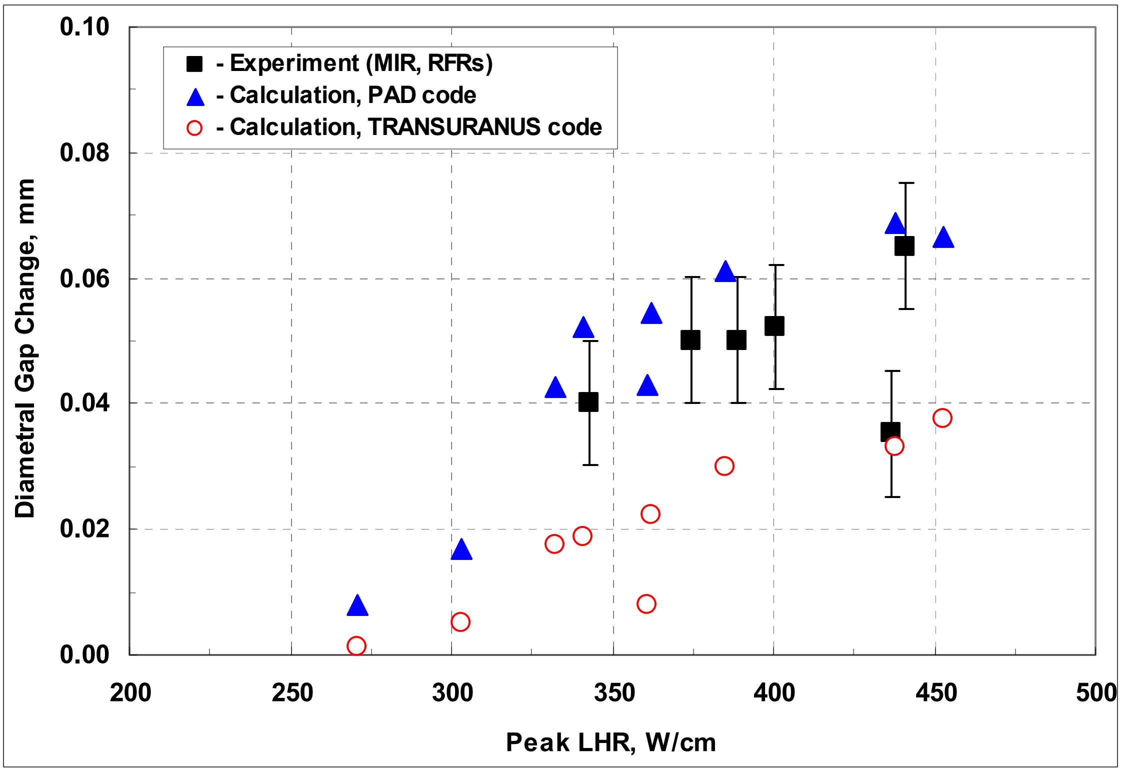


Figure 1-12. Dependence of pellet-to-cladding diametral gap change after power ramp versus maximum LHR measured for refabricated WWER-440 rods and calculated by PAD and TRANSURANUS codes

- <u>Ramp test</u>

The gap size measurements of refabricated WWER-440 FRs performed after power ramp tests showed an increase in the diametral gap. Figure 1-12 demonstrates the variation of the diametral gap change measured in the range of the maximum LHR as function of the peak LHR. The PAD and TU calculation results of the FA-198 & 222 RFRs are displayed in the same graph. The figure allows making the following conclusions:

- in the LHR range of (34 ÷ 45) kW/m the measured gap values overestimate the TRANSURANUS results. The deviation between the measured and the calculated values is about 20 µm;
- the PAD results correlate well with the experimental data and lie within the measurement uncertainties. Thus, at the power ramp level of ~45 kW/m, the calculated value of gap increment is 70 µm versus the measured 65 µm.

1.4.6 Cladding Elongation

In accordance with the IFPE database [6], the initial cladding length of the FA-198 & 222 FRs is known as 2553.33 mm, on the average, with a standard deviation of 1.44 mm. The rod length measurement error is ± 0.3 mm. Thus, ± 1.73 mm or 0.07% can be considered as the actual measurement uncertainty. Taking it into account, the comparison of the measured and the calculated data of WWER-440 rod elongation, which is presented in Figure 1-13, allows making the following conclusions:

- *TRANSURANUS code*

The code calculated results agree well with the measured ones only for the rods having the maximum BU less than ~57.5GWD/MTU. Though, during the analysis the model of fuel pellet-to-cladding interaction and sliding friction was taken into account, the TU-WWER results trend to increase with burnup growth. The code-predicted rod elongation at the maximum fuel burnup of 65 GWD/MTU is about 0.57%.

- *PAD code*

The code results do not show the systematical over- or underestimation for the WWER-440 FRs examined. Nevertheless, the PAD calculated data show the tendency of cladding increase with rod burnup growth. The deviation between the measured and the calculated rod elongation averaged over all FRs tested has the minimum in the case, when the model parameters for the UB fast neutron fluence are utilized (see Figure 1-13). For these parameters, the expected value of WWER-440 rod growth at the maximum fuel burnup of 65 GWD/MTU is ~0.5%.

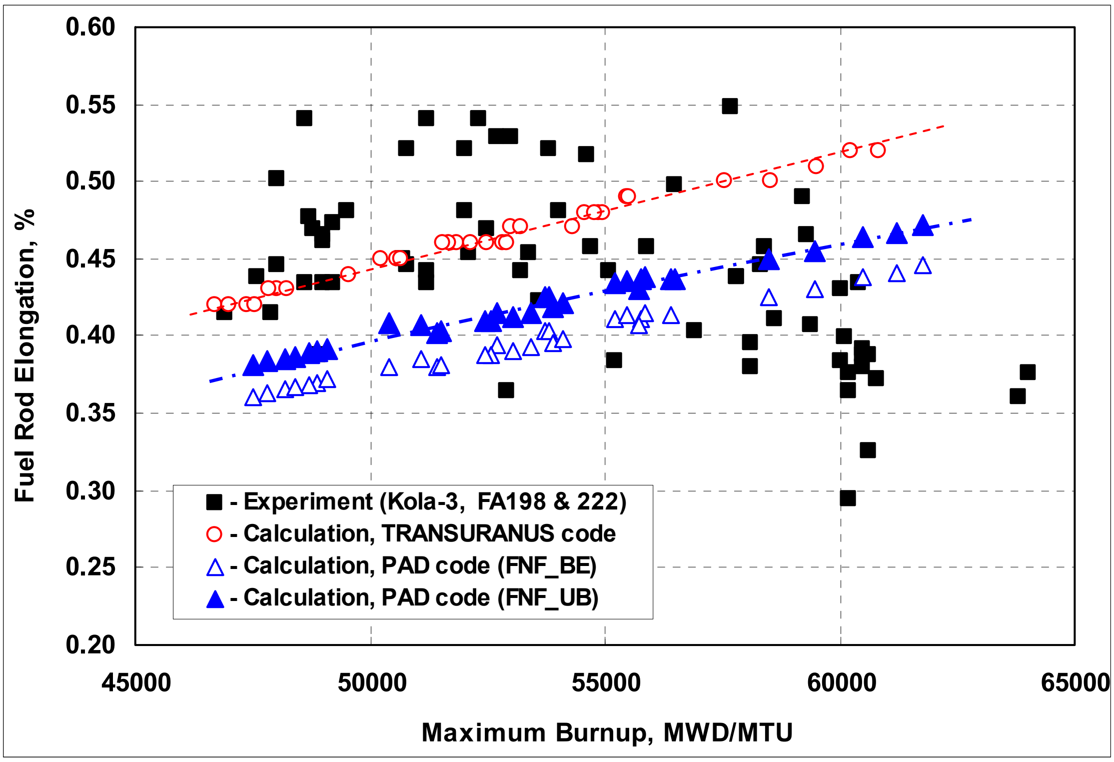


Figure 1-13. Dependence of fuel rod elongation versus maximum burnup measured for FA-198 & 222 FRs after base irradiation and predicted by TRANSURANUS and PAD codes

1.4.7 Fission Gas Release and Rod Internal Pressure

The investigation results performed for Russian fuel types fabricated using different technology parameters showed [8] that the initial pore microstructure of fuel causes significant impact on the pellet densification and swelling. For the "old" fuel type, which was used for manufacturing of WWER-440/1000 FRs (utilized in FA-198 & 222 and FA-E0325 & ED4108 etc. [17]), the maximum fuel densification rate was registered. In the initial state, the microstructure of this fuel type is presented by pores of different sizes and volume density in the grain volume. Obviously, during the operation the pores evolution will course for different ways across the fuel pellet. Certainly, the pores efficiency, as traps for the fission gas atoms, and the temperature migration of theses gas pores are different. That, in turn, can significantly impact FGR from the pellet volume during the normal operation conditions and the power transients.

- Steady-State Operation

To estimate the prediction capability of the PAD and the TRANSURANUS FGR models the deviation between the measured and the calculated values of fission gas release and RIP for the FA-198 & 222 FRs tested was built. These deviations versus the rod burnup are presented in Figure 1-14. The figure shows:

➢ *TRANSURANUS code*

- as the standard FGR model is used, the calculated data are slightly underestimated at the burnup range of (40 ÷ 45) GWD/MTU. The underestimation increases with rod burnup growth. The average deviation of FGR is 0.81 % (S.D. = 0.71 %).

  The TU simulation results obtained during the analysis of WWER-1000 FRs [11] showed that a decrease in the threshold burnup for the high-burnup structure FGR-model allows to increase gas release as well as rod internal pressure in the FRs tested. Similar results were obtained in [18], when WWER-440 rods were analyzed by means of TU-WWER fuel rod models.

  The sensitivity analysis performed for different model parameters, which could impact steady-state fission gas release, showed that better agreement with the experimental data is reached, when the threshold burnup of 75000 MWD/MTU (code parameter *ThBrnp*) is utilized. The results of FGR after the base irradiation calculated for the WWER-440 RFRs demonstrate this effect (see Table 1-3);

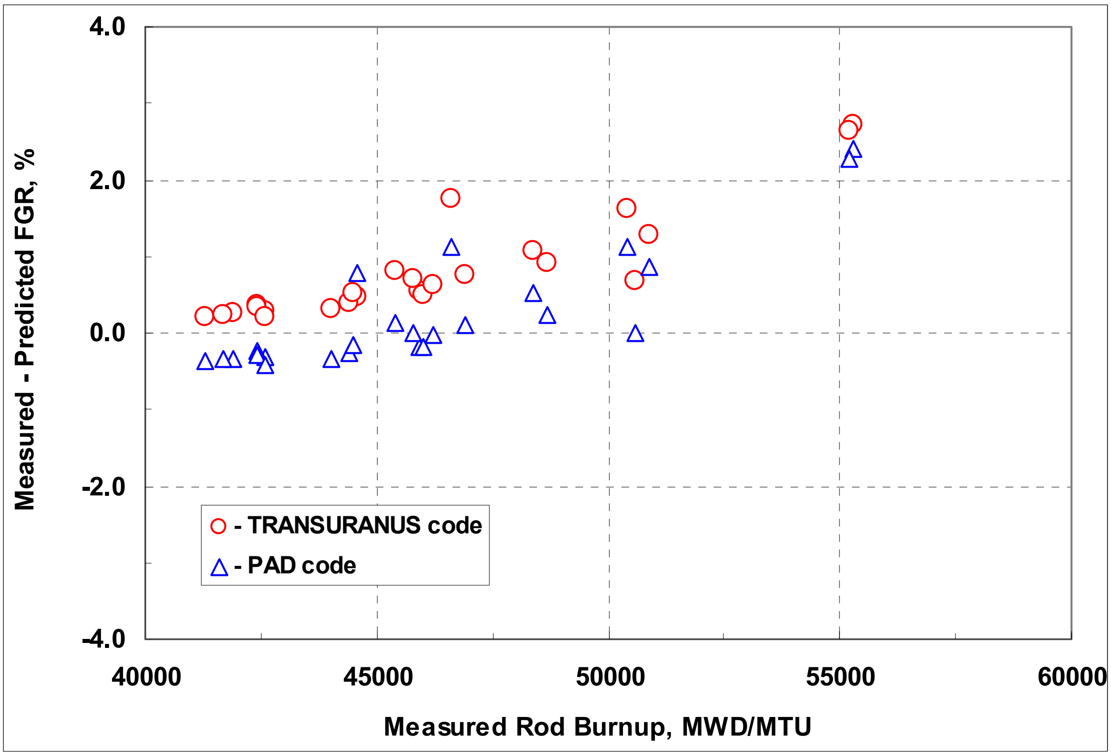
- the calculated RIPs follow the trend of FGR data. The average deviation of RIP (EOL, normal condition) is -0.28 MPa (S.D. = 0.13 MPa).

➢ *PAD code*

- as the initial (best estimate) parameters of FGR model are used, the calculated data are in satisfactory agreement with the measured ones obtained for the rod burn-ups ranged in (40 ÷ 50) GWD/MTU and are underestimated at the rod BU of ~55 GWD/MTU. The average deviation of FGR is 0.24 % (S.D. = 0.78 %);

- for the most FRs tested the deviation between the measured and the calculated RIPs at the normal condition lies within the manufacturing tolerance, ±0.1 MPa (see Figure 1-14).

  The sensitivity analysis performed for the different fuel swelling and rod growth models revealed that the predicted RIPs agree well with the measured ones. The average deviation of RIP varied from -0.05 MPa (S.D. = 0.24 MPa) to +0.06 MPa (S.D. = 0.12 MPa).



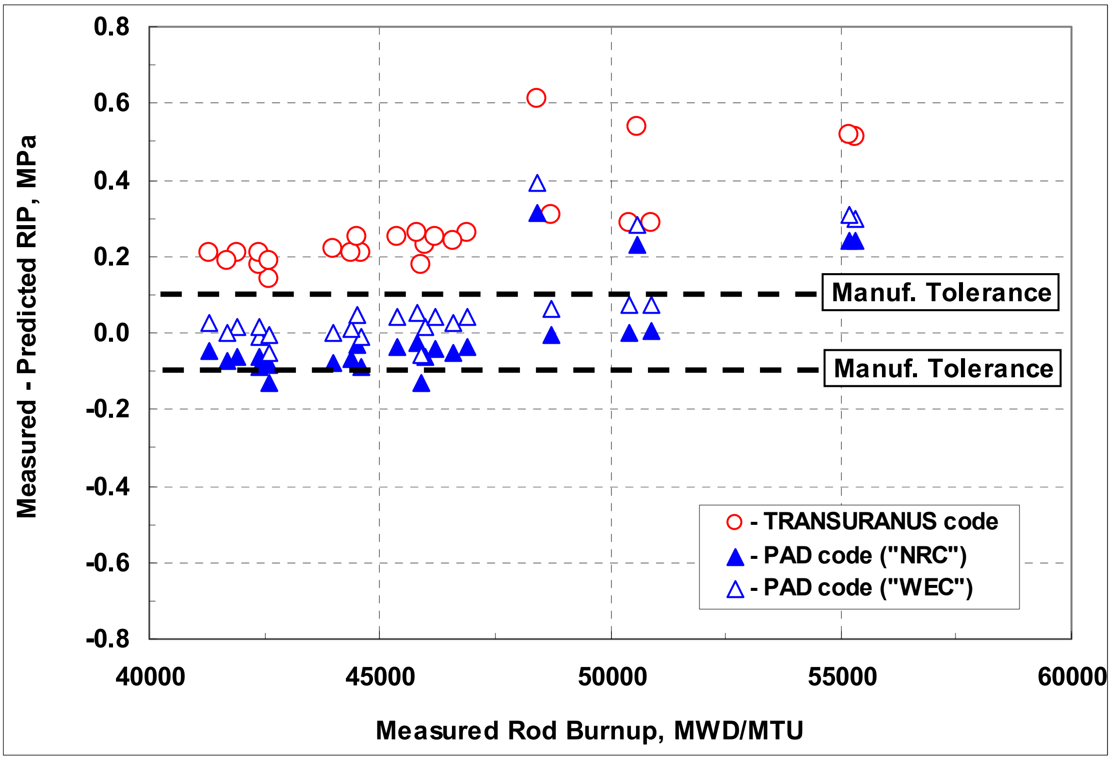


Figure 1-14. Deviation (measured minus code predicted) of fission gas release and rod internal pressure of FA-198 & 222 fuel rods versus rod average burnup

- <u>Ramp test</u>

The measured and the calculated values of FGR after the power ramp for the refabricated FA-198 & 222 FRs are listed in Tables 1-3 and 1-4, for the TRANSURANUS and the PAD code, respectively.

The measured and the calculated RIP data the ramp test conditions for RFRs №41 and №48 are presented in Figure 1-15.

Below is the analysis of the obtained data.

➢ *TRANSURANUS code*

- as the standard FGR model is used, the calculated data differ from the measured ones and are significantly underestimated (see Table 1-3).

  In order to estimate the predicted capability of different model options, the model of the intergranular fission gas release was utilized. This model describes the release of gas atoms from the grain boundaries to the FR free volume, when the specific saturation concentration of gas atoms on boundaries is reached. The following model options were tested: (i) "standard", with the constant saturation concentration (see 1.3), (ii) "ramp release", which considers the burst gas release due to micro-cracking under ramp conditions, (iii) "the maximum release", which allows to release immediately all fission gas atoms reached to the grain boundaries. The calculated results demonstrate that for the "maximum release" option the predicted FGR is still far from the experiment. It is should be noted, that similar results were obtained by authors of [19] and assumed that "*there is not sufficient fission gas 'arriving' at the grain boundaries (simulated by diffusion)*";

- concerning the pin pressure. As it follows from the "FGR-1" test, the pressure increase is the same as the rod power growth (see Figure 1-3 and Figure 1-15). As was found, the measured pressure in RFR №48 (BU=60.5 GWD/MTU) reaches the RIP of RFR №41 (BU = 48.9 GWD/MTU), when the rod heat power increases to 20 kW/m and ~25 kW/m, correspondingly. For this power conditions simulated, the calculated FGR are significantly below the measured ones. The deviation between the measured and the calculated RIP is 3.8 MPa (rod №41) and 7.5 MPa (rod №48).

  At the second stage of ramp test, when the peak LHR jumps to ~43 kW/m, the calculated RIP increases to 11.7 MPa. However, the predicted value is 5 MPa lower than the measured one.

➢ *PAD code*

- for the initial parameters of FGR model and $\lambda(L25)$-correlation for UO_2 thermal conductivity, the calculated FGR data are underestimated for the power transient conditions of "FGR-1" and "RAMP" tests. At the same time, the calculated FGR is overestimated for Rod №52 ("FGR-2" test, Table 1-4). The predicted FGR for Rods №51 and №50 is ~59 % and ~45 %, respectively. The deviation of transient FGR averaged over all rods examined is 10.9 % (S.D. = 8.7 %).

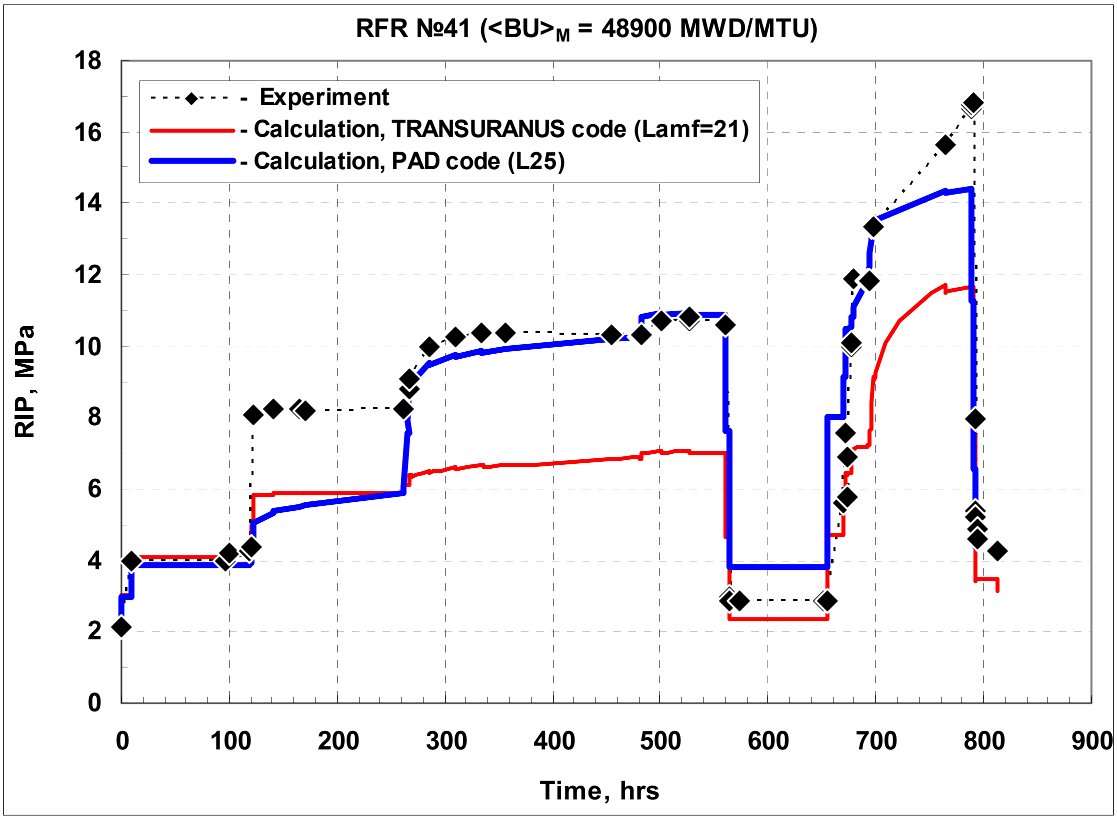
  The increment of fission gas release, ΔFGR, due to rod power increase by 6 % (measurement error) is not significant to reduce a difference between the measured and the calculated FGR observed for the RFRs from "FGR-1" and "RAMP" tests. Thus, for the rods with the initial BU of ~50 GWD/MTU the ΔFGR is (5 ±0.5) % and this value is ~4.1 % for the rods with the initial BU of ~60 GWD/MTU.

Table 1-3. Measured and TRANSURANUS code predicted fission gas release at steady-state operation and power transients for refabricated fuel rod of FA-198 and FA-222 assemblies.

RFR Number (FA #)	**Maximum burnup after base irradiation, GWD/MTU**		**FGR after base irradiation, %**			**FGR after test irradiation, %**			
			TRANSURANUS calculation with different FGR-model options						**Puncturing Results**
	Measured	**Calculated**	**Standard & *ThBrnp*= 85000**	**Standard & *ThBrnp*= 75000**	**Ramp release**	**Standard & *ThBrnp*= 85000**	**Ramp release**	**Max. release**	
“FGR-1” test									
41 (198)	48.9	46.57	0.29	0.40	0.57	15.5	19.1	***19.5***	**47.5**
32 (222)	60.2	57.53	0.95	2.10	1.43	5.4	8.0	***8.2***	**47.0**
48 (222)	60.5	58.42	1.01	2.35	1.49	5.5	8.3	***8.6***	**50.0**
“FGR-2” test									
51 (198)	49.5	47.96	0.30	0.70	0.61	26.9	32.5	***33.1***	**-**
52 (222)	58.0	56.93	0.90	1.89	1.38	18.3	22.7	***23.3***	**48.4**
50 (222)	58.4	57.07	0.91	1.93	1.39	10.9	15.2	***15.7***	**-**
“RAMP” test									
33 (198)	50.8	46.90	0.29	0.49	0.58	0.92	2.44	***2.64***	**31.3**
37 (222)	60.1	57.20	0.92	2.01	1.39	0.98	1.98	***2.02***	**16.9**
38 (222)	60.2	57.43	0.94	2.07	1.41	0.98	1.54	***1.59***	**19.6**

Table 1-4. Measured and PAD code predicted fission gas release at steady-state operation and power transients for refabricated fuel rod of FA-198 and FA-222 assemblies.

RFR Number (FA #)	Maximum burnup after base irradiation, GWD/MTU		FGR after base irradiation, %			FGR after test irradiation, %			
			PAD calculation with different code model parameters						Puncturing Results
	Measured	Calculated	Standard FGR; λ(*L25*)	Standard FGR; λ(*L30*)	Modified FGR; λ(*L25*)	Standard FGR; λ(*L25*)	Standard FGR; λ(*L30*)	Modified FGR; λ(*L25*)	
"FGR-1" test									
41 (198)	48.9	48.47	0.97	0.97	0.97	***39.4***	42.2	41.5	**47.5**
32 (222)	60.2	60.03	1.43	1.43	2.63	***26.5***	30.2	34.0	**47.0**
48 (222)	60.5	61.52	1.49	1.49	3.10	***30.8***	35.3	39.2	**50.0**
"FGR-2" test									
51 (198)	49.5	49.82	1.03	1.03	1.03	***59.4***	61.9	62.0	**-**
52 (222)	58.0	59.68	1.42	1.42	2.51	***53.2***	57.1	59.1	**48.4**
50 (222)	58.4	59.78	1.42	1.42	2.53	***45.2***	50.0	50.3	**-**
"RAMP" test									
33 (198)	50.8	49.23	0.99	0.99	0.99	***18.3***	21.7	18.6	**31.3**
37 (222)	60.1	60.00	1.42	1.42	2.63	***10.9***	14.3	12.5	**16.9**
38 (222)	60.2	60.27	1.44	1.44	2.77	***4.8***	7.4	6.1	**19.6**



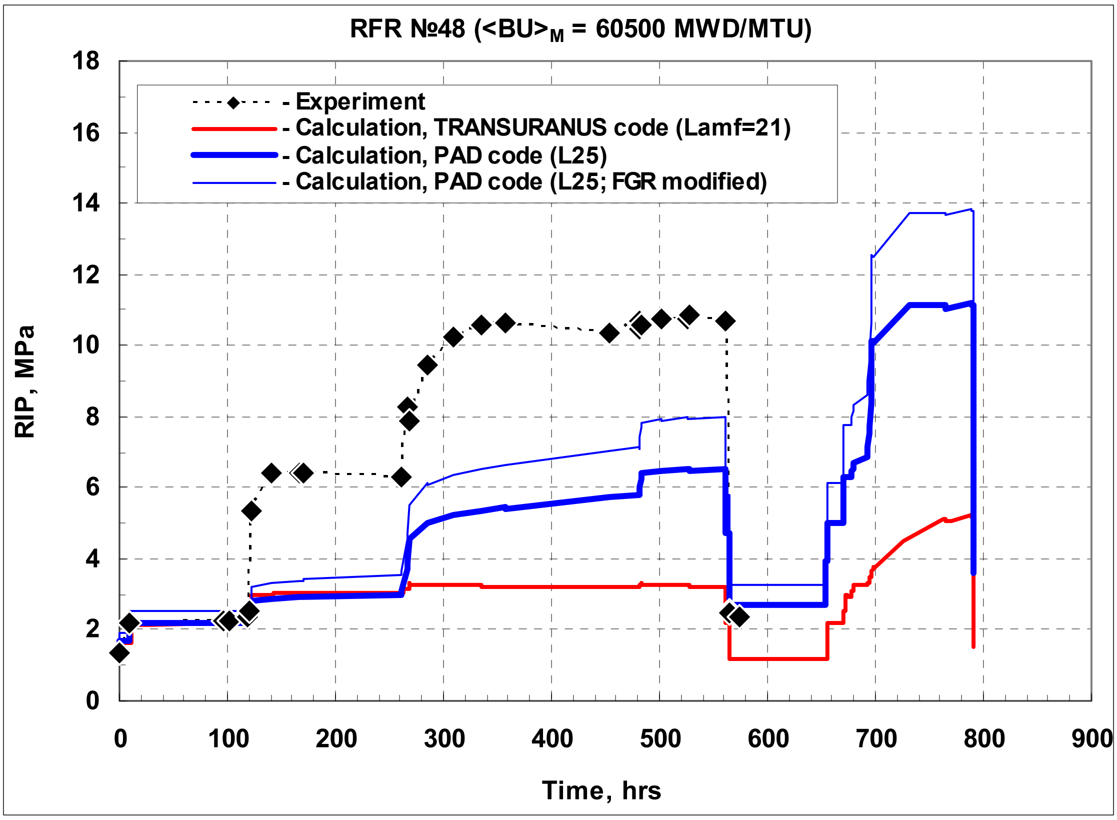


Figure 1-15. Measured and code-predicted rod internal pressure variation for refabricated fuel rods №41 and №48 during power ramp test "FGR-1" carried out in reactor MIR

To estimate the effect of some model options' impact on transient FGR, the code sensitivity analysis was carried out. The analysis results showed:

(i) the reduction of gas atom diffusion length to grain boundaries due to the decrease in the grain size increases FGR. The calculated increment of FGR, when the grain size was reduced from 7 μm to 6 μm (see Table 1-1), varied from 1% to 2.9% for the FRFs examined;

(ii) the reduction in thermal conductivity of UO_2 due to burnup effect leads to an increase in the fuel temperature, that in turn intensifies a FGR from the fuel. When $\lambda(L30)$-correlation is applied, the increment of FGR is varied from 2.5 % to 4.8 % (see Table 1-4). The maximum $\Delta FGR \approx 4.5\%$ is realized for rods №48 and №50 having the maximum burnup, ~60 GWD/MTU.

For the case considered, the deviation of FGR averaged over all rods examined is 7.4 % (S.D. = 8.7 %);

(iii) as it was found in [11, 12] during the rod analysis of Russian WWER-1000 fuel by means of PAD code, a decrease in the threshold BU & T (FGR model parameters) decreases the difference between the measured and the calculated FGR at steady-state operation for the rod burn-up exceeding 50 GWD/MTU. The same is also observed for WWER-440 fuel (see Table 1-4, column - "modified FGR"). When these model parameters are utilized, the appreciable increase in FGR is observed. The maximum increment of FGR is (5÷8.5) % and is realized for the high-burnup rods (~60 GWD/MTU) subjected to 25 kW/m (and higher) power ramp.

For the case considered, the deviation of FGR averaged over all rods examined is 7.0 % (S.D. = 8.6 %).

As mentioned before, the initial complex pore structure of the considered WWER fuel can impact the formation of gas pore structures, which influence fission gas atoms release during transients. This assumption is partly confirmed by the experimental data of the pin pressure variation in rods №41 (48.9 GWD/MTU) and №48 (60 GWD/MTU) at the first stage of the ramp test (see Figure 1-15) and by the differences of fuel microstructure for these rods after the ramp. In accordance with the measured data [9], for Rod №41 the area of fuel pellet subjected to gas swelling is 1.8 times higher than the one observed in the fuel pellet of Rod №48;

- concerning the pin pressure. The variation of rod internal pressure at power follows fission gas release. For Rod №41, during the first sub-stage of the test the calculated RIP is underestimated and is close to the TRANSURANUS calculated data (see Figure 1-14). At the second sub-stage of power transient the calculated RIP data agree well with the measured ones. For the last stage, the calculated RIPs are close to the measured ones, when the rod power increases. However, the calculated pin pressure rate at the maximum heat power is less than it is measured. So, at the rod power of 43 kW/m, the deviation between the measured and calculated RIP is 2.35 MPa.

  The pin pressure calculated for the rod №48 is less than the measured one. At the rod power of 19.5 kW/m, the deviation of RIP is 4.3 MPa. The deviation increases to 2.8 MPa, when the modified FGR model parameters are used (see Figure 1-15).

2. PWR (AREVA) Fuel Rod Performance Analyses

The PAD and TRANSIRANUS analysis results of PWR fuel rod performances in the burnup range of (37-81.5) GWD/MTU are presented in this chapter. This case, referred as AREVA idealized case, was a priority case for FUMEX-III project and is based on measurements for fuel rods operated for 3, 4 and 7 cycles in a commercial French PWR reactor.

2.1 PWR Fuel Rod Specification

The design characteristics of PWR fuel rods manufactured by AREVA are taken from [3] and presented in Table 2-1.

Table 2-1. PWR Fuel Rod Specifications

Parameter	Value
Fuel Rod	
Fuel stack length, mm	3650
Plenum volume, cm^3	8.04
Fill gas composition	He (100%)
Backfill gas pressure, MPa	1.6
Fuel Rod Cladding	
Material type	Zr-4 (stress-relieved)
Tube outer diameter, mm	9.50
Tube inner diameter, mm	8.25
Fuel Pellets	
Material type	UO_2
Enrichment, wt.% ^{235}U	4.5
Pellet type	Solid
Pellet height, mm	13.25
Pellet outer diameter, mm	8.085
Chamfer height, mm	0.270
Chamfer width, mm	0.543
Dishing depth, mm	0.310
Dishing radius, mm	3.00
Dishing volume (2-sides), mm^3	8.8
Pellet grain size, μm	~10
Fuel density, %T.D.	95.0

2.2 PWR Operational Conditions

Three fuel rods were operated in 3, 4 and 7 cycles in a commercial French PWR reactor. Those rods experienced similar power histories, i.e. the 3 and 4 cycle rods match very closely the first 3 and 4 cycles, respectively, from the 7 cycle rod. The thermal-hydraulic parameters of PWR core were within the nominal operating conditions – the coolant inlet temperature is 286 ^{0}C; coolant pressure is 15.5 MPa.

The rod power history of the idealized case is available in [3]. Variation of the rod average and the maximum LHR during the normal operation is shown in Figure 2-1.

The data of rod average burnup and the expected FGR after 3^d, 4th and 7th cycle operation are listed in Table 2-2.

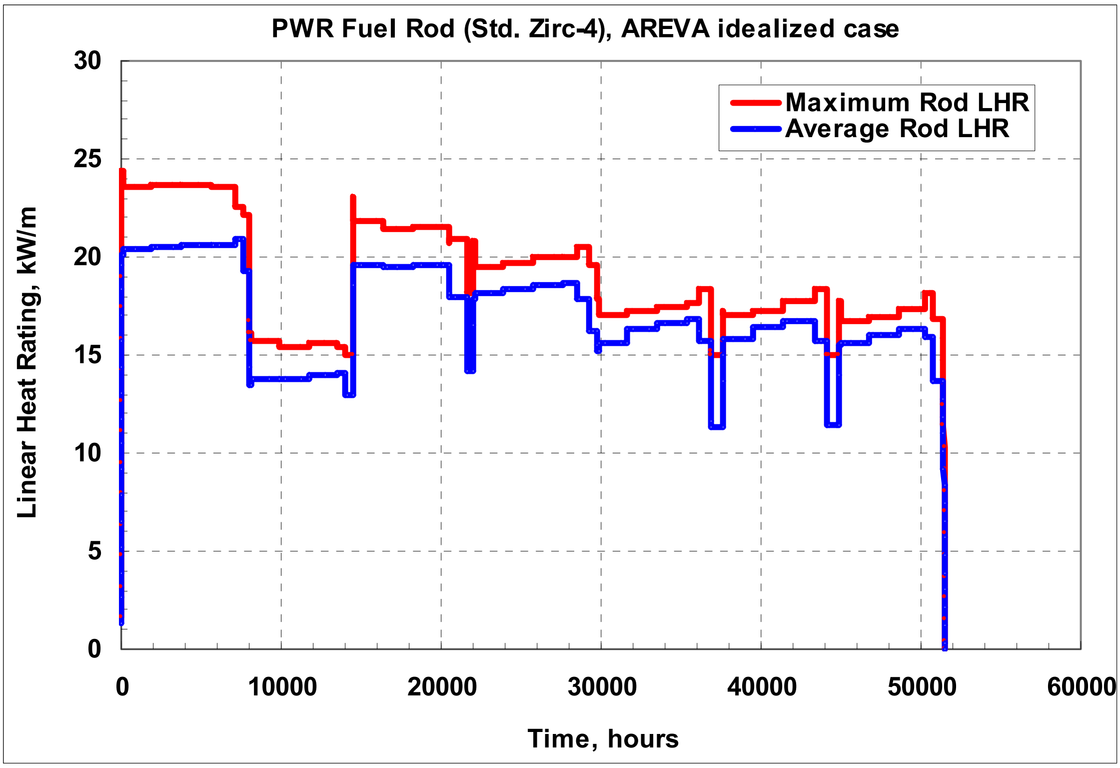


Figure 2-1. Dependence of maximum and rod average linear heat rates of PWR fuel rod versus time operation

Table 2-2. PWR fuel rod burn-ups and expected FGR

End of cycle	Time operation, days	Burnup, GWD/MTU	Expected FGR value, %
3	916.4	36.6	0.5 (+0.5/-0.2)*
4	1239.1	49.7	1.9 (+1.0/-0.7)
7	2141.9	81.5	9.0 (+2.5/-2.0)

* The given FGR uncertainties allow for measurement, fabrication and irradiation uncertainties.

2.3 Description of PAD and TRANSURANUS Fuel Rod Models

➢ *PAD code*

The PAD-4.0 version 10.5.2 fuel rod model with standard Zr-4 alloy cladding was used as a basis during the current FR analysis.

The corrosion model for standard Zr-4 alloy cladding developed for the rod burnup of 62 GWD/NTU utilizes the cooling chemistry conditions during the irradiation. The validation calculations performed using the PWR 16x16 LTA data showed a good agreement between the PAD predicted and the measured oxide thickness for the rods burned in the range of 55-60 GWD/MTU [20]. Since for the given analysis the data of cooling chemistry conditions per each cycle operation are not available, the code parameter of corrosion rate was used. This parameter allows the simulating of cladding oxidation in the reasonable thickness range of (70-100) µm, when the rod burnup grows to ~81 GWD/MTU.

The best estimate parameters of FGR model and "*WEC*" fuel swelling and densification model were used as the basis.

The thermal conductivity correlation for UO_2, λ(*L25*), which takes into account the effect degradation during the fuel burnup growth, was used as the initial one. This correlation provides a good agreement with experimental data of FCT measured for WWER rods and with the FCT calculated by others λ-correlations presented in TRANSURANUS code (*Lamf*=21; 25).

The profiles of radial power distribution across the fuel pellet in dependence on fuel burnup calculated using the TUBRNP module of TRANSURANUS code were utilized in PAD code as basic ones during the analysis.

➢ *TRANSURANUS code*

The code TRANSURANUS v1m1j09 (TU), which takes into account the relevant phenomena occurring in the PWR fuel (code parameter "*PWR*") and the specific features of the cladding made from Zirc-4 alloy (code parameter "*ZIR*"), was used in this analysis.

The calculations have been carried out coherently with the power history of the PWR rod examined and using the design characteristics from Table 2-2. The value of fuel pore removable during sintering of 1.11 % and burnup at which sintering has stopped of 5 GWD/MTU were used based on the previous analysis results of PWR fuel designs with initial fuel density varied in the range of (94.4 ÷ 95.3) % T.D. [20].

Below, the following TU fuel rod models were used -

- the simple empirical model of fuel densification and swelling [13];
- the formation of fuel cracks, which are considered an adder to the rod free volume. The BE parameter of fuel crack model is applied;
- the fuel pellet-to-cladding interaction with the BE static and sliding friction coefficients and without accounting for slip for axial PCMI;
- the grain growth (model of Ainscough and Olsen);
- the intragranular fission gas release (*URGAS* model with the BE thermal and athermal diffusion coefficients, proposed by Hj. Matzke);
- the intragranular fission gas release with the BE grain boundary gas saturation concentration of 1E-4 mol/m^2;

- the FGR from the fuel high-burnup structure (*HBS* model with the BE threshold burnup of 85000 MWD/MTU, as the basis;
- the cladding material properties for stress relieved Zirc-4 (parameter *ModClad*(4)=18);
- the standard LWR settings for UO_2 fuel properties with the fuel thermal conductivity according to Harding and Martin correlation for UO_2 (*Lamf* =21).

2.4 Simulation Results of PWR Fuel Rod by PAD and TRANSURANUS Codes

2.4.1 Fuel Burnup

The PAD and TRANSURANUS calculated results of the rod average burn-up growth during 7 cycles operation are presented in Figure 2-2. The measured data are shown in the same figure.

The obtained data evidence:

- there is a good agreement between the measured and the codes predicted rod burnup in the range up to 81.5 GWD/MTU;
- at the EOL the maximum deviation between the expected and the predicted rod burnup is observed for TU code and is 3.2%.

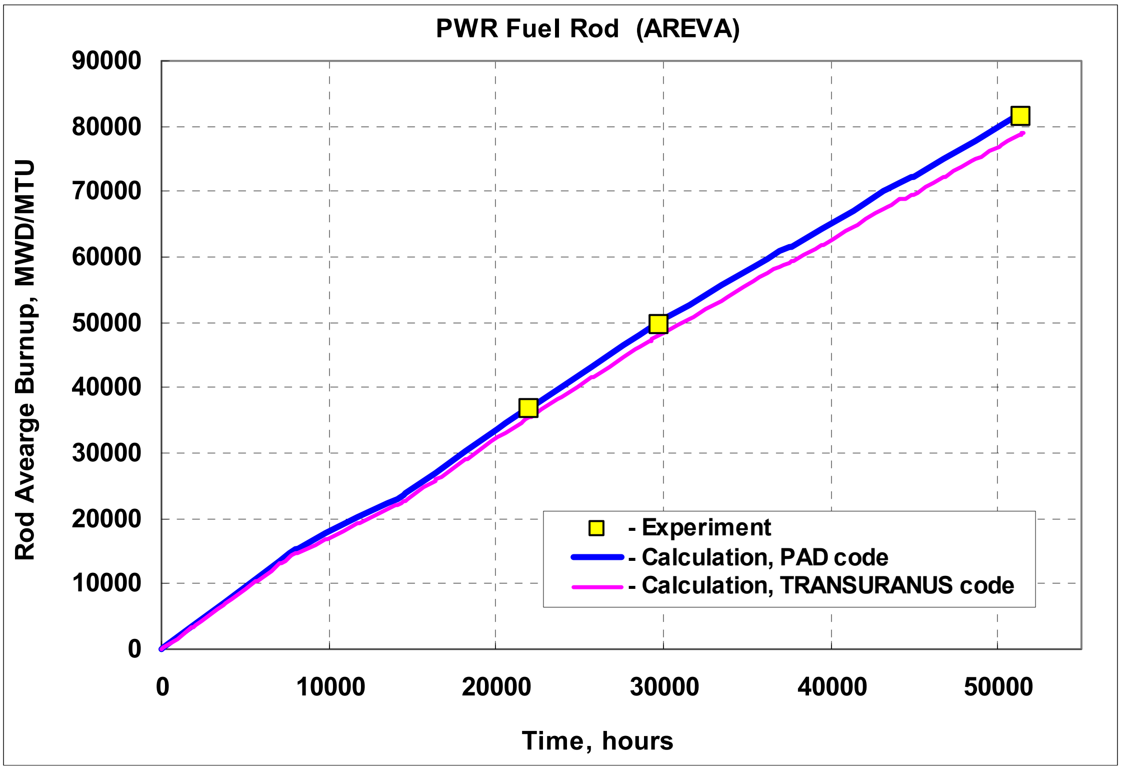


Figure 2-2. Dependence of rod average burnup versus time operation measured for PWR rod and predicted by TRANSURANUS and PAD codes

2.4.2 Steady-state fission gas release

The code prediction results of steady-state FGR variation during 7 cycles operation are presented in Figure 2-3. The measured (expected) data are shown in the same figure.

On the whole, the dependence of FGR vs. rod burnup calculated by both codes passes within the expected FGR data.

➢ *TRANSURANUS code*

The current FGR-model used, which accounts the gas release form HBS, provides a satisfactory agreement with the experimental data. At the rod <BU> of 78622 MWD/MTU calculated at EOL the value of fission gas release predicted by the code is ~9.7 %.

➢ *PAD code*

For the initial parameters of FGR model and λ(*L25*)-correlation for UO_2 thermal conductivity, the calculated FGR data lie at the upper bound of the measured ones.

In order to estimate the influence of code thermal model parameters on the fission gas release the sensitivity analysis was performed. The model parameters for λ-correlation of UO_2 thermal conductivity and cladding corrosion were varied. The calculation results showed:

- when the λ(*L20*)-correlation is used, the predicted FGR data are in good agreement with the measured ones (see Figure 2-3);
- the increase in cladding oxide thickness results in an increased FGR due to the fuel temperature growth. The increment of FGR is ~2.5%, when the peak oxide thickness expected at EOL grows from 70 µm to 120 µm.

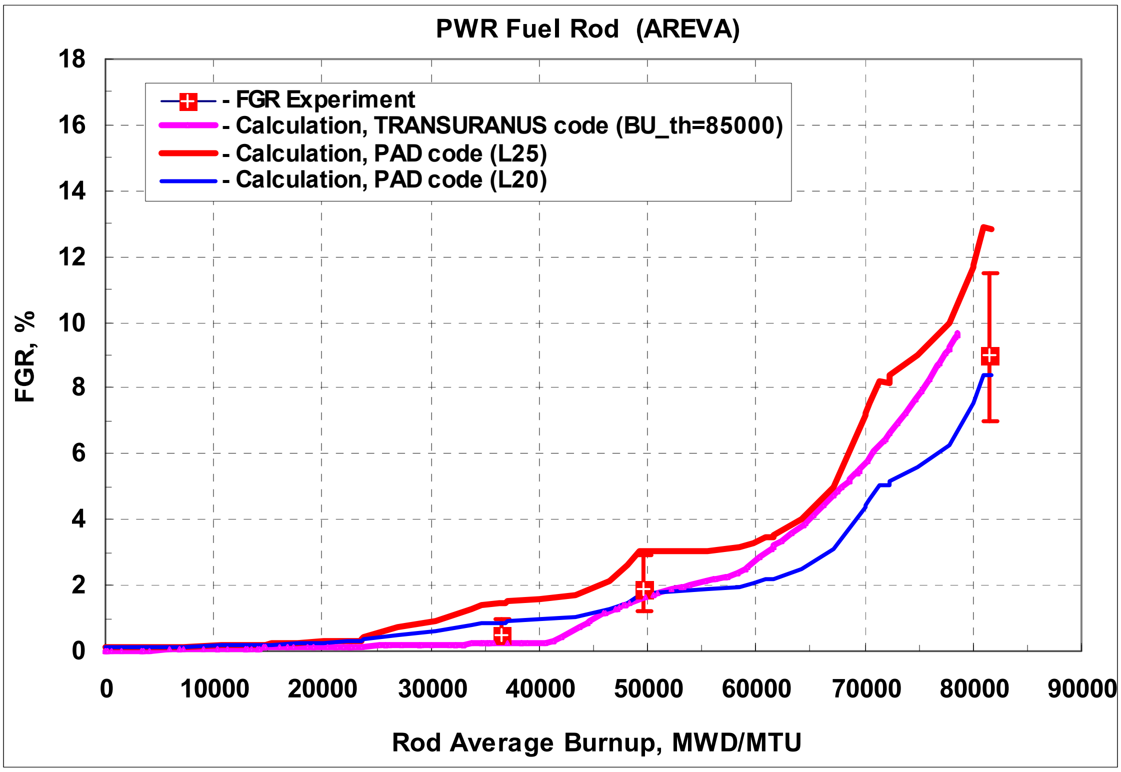


Figure 2-3. Dependence of FGR versus rod burnup measured for PWR rod and predicted by TRANSURANUS and PAD codes

3. Conclusion

The prediction capability of the TRANSURANUS and PAD codes for evaluating PWR and Russian WWER-440 fuel rod behavior under steady-state operation conditions and power transients has been assessed on the basis of the experimental data presented in the IFPE database of the OECD/NEA.

The analyses, which have focused on some integral quantities, have firstly pointed out that the WWER rod average and the maximum burn-ups calculated by both codes lie within the measured uncertainties. For the PWR (AREVA) rod tested, the PAD calculated burn-ups are in good agreement with the measured ones, while the TU results are continuously underestimated by ~3.5 % on the average in the range of (37 ÷ 81.5) GWD/MTU.

As a whole, the TRANSURANUS calculated results are in satisfactory agreement with the experimental data for WWER-440 FRs normally operated during 4÷5 cycles to the rod burn-ups ranged from 41.3 to 55.4 GWD/MTU. As for PAD code with adjusted model parameters for Russian WWER FR design, the calculated data of cladding creepdown (cladding OD change), pellet-to-cladding gap, FGR and RIP show better agreement with the measured ones.

As concerns the fuel temperature of WWER rods at steady-state operation, the maximum fuel centerline temperature calculated by both codes is reached during the 1st cycle and at BOC-2 and the maximum FCT stays below 1000 ^{0}C. Comparison of the maximum FCTs calculated by the codes for the high-burnup rod showed that the TU correlations for UO_2 thermal conductivity (λ_{21}, λ_{25}) versus PAD thermal model with $\lambda(L25)$-correlation provide higher temperature in a fuel at the BOL operation. For "fresh" fuel, the difference is ~100 ^{0}C. In the high-burnup range of (50 ÷ 60) GWD/MTU the difference of FCT is ~40 ^{0}C and -25 ^{0}C versus λ_{21}- and λ_{25}-correlation respectively.

There is not a good correlation between the measured and the calculated rod performances for the refabricated WWER-440 fuel rods during the transients. As found, for Rod №51 (BU = 49.5 GWD/MTU) the fuel centerline temperatures during the power ramp calculated by both codes are in good agreement with the measured ones, while for Rod №50 with higher burnup (58.4 GWD/MTU) the FCTs recorded are higher than those predicted by the codes. For all RFRs tested, the measured fission gas release and RIP at power are significantly underestimated by the TRANSURANUS code, even though some models for extreme FGR were used. The PAD results for FGR are under- and overestimated. However, the deviation between the measured and the predicted FGR is underestimated by 10.9 % on the average. The deviation decreases to 7 %, when the modified parameters (threshold burnup and temperature) for the PAD high-temperature FGR model are utilized.

The TU calculation results for steady-state FGR are in good agreement with the measured data for the PWR (AREVA) fuel rod burned up to 81.5 GWD/MTU. The PAD calculation results, which take into account the effect of cladding oxide thickness on fuel temperature, are slightly overestimated, when the thermal model with $\lambda(L25)$-correlation for UO_2 and the initial FGR model parameters are utilized.

To sum up, the TRANSURANUS-WWER FR models and the existing PAD FR models with the modified parameters of cladding corrosion rate, the cladding creep rate, the UO_2 thermal conductivity, threshold burn-up and temperature for the high temperature fission gas release showed satisfactory agreement with the measured data for Russian WWER fuel rods normally operated up to 60 GWD/MTU. For the WWER fuel type tested, the observed difference between the calculated and the measured transient FGR points out that the existing FGR models can be modified to account for the gas pores thermal stability, as an additional source of fission gas release under the transient conditions.

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