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Best estimate analysis of processes in RBMK fuel rods during the operation cycle

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1. Introduction

Ignalina NPP is the only nuclear power plant in Lithuania which is comprised of two units, commissioned in 1983 and 1987. Both units are equipped with channel-type graphite-moderated boiling water reactors RBMK-1500. Ignalina NPP Unit 1 was shutdown for decommissioning at the end of 2004 and unit 2 was shutdown at the end of 2009. After final showdown of reactor the fuel assemblies from the reactor are transferred to the spent fuel pool where they will be kept at least for 5 years. Thus the state of the fuel rods (intactness of cladding, residual stresses in the cladding and fuel pellets, gap between the cladding and pellets and etc.) is very important because fuel rod cladding is one of the safety barriers. For this purpose the deterministic analysis of processes in the fuel rods are performed.

In the beginning, Ignalina NPP operated with 2.0% U^{235} enrichment fuel. Before the final shutdown, the fuel of 2.4%, 2.6% and 2.8% U^{235} enrichment with burnable erbium absorber was used at Ignalina NPP. The amount of erbium depends on the enrichment of fuel: for 2.4% enrichment there is 0.41% of erbium, for 2.6% enrichment – 0.5% and for 2.8% enrichment – 0.6%. Nuclear fuel is compressed to pellets of 11.46 mm diameter and 15 mm height [1]. The shape of the pellet is adapted to the intensive, high-temperature operating mode. The 2 mm diameter hole through the axis of the pellet reduces temperature at the centre of the pellet. In order to decrease the neutron escape from the reactor core in the axial direction, screen pellets of 0.7% U^{235} enrichment are included at the end close to the gag of the fuel rod.

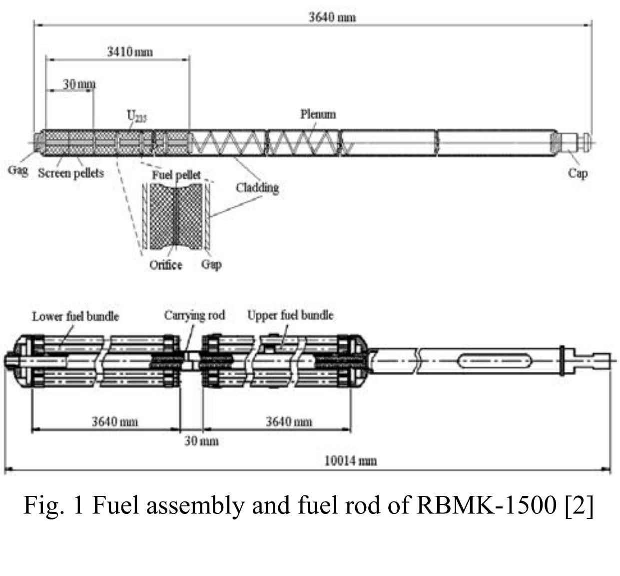

Fig. 1 Fuel assembly and fuel rod of RBMK-1500 [2]

The pellets are placed into a cladding with the outside diameter of 13.6 mm, wall thickness of 0.9 mm and active length of 3.41 m (Fig. 1). The fuel cladding is made of a zirconium and niobium alloy. Mechanical properties of this zirconium alloy is presented in [3]). In RBMK reactor the fuel assembly is fit into a circular fuel channel with the inside diameter of 80 mm and core height of 7 m. In order to achieve the required height, the RBMK fuel assembly consists of two fuel bundles placed one above the other (Fig. 1). Each fuel bundle includes 18 fuel rods placed in two circles around the carrying rod.

This paper presents the information about the model development using FEMAXI-6 code and identification of fuel rods status after the normal operation.

2. Development of fuel rod model

Lithuanian Energy Institute (LEI) uses FEMAXI-6 code for modeling of the processes in the fuel rods of Ignalina NPP [4]. FEMAXI-6 code is designed for vessel type reactors. The materials of fuel and cladding and design of fuel rods in RBMK-1500 are different comparing to vessel type light water reactors. These differences are evaluated during the development of the model.

The FEMAXI-6 code was used because this code produces more detailed calculations and predicts thermal and mechanical behaviour of a light water reactor fuel rod during normal operation and transient conditions. Additionally, the code persists simply and is an open-source, which is a significant reason for choosing this code. FEMAXI-6 code can analyze the integral behaviour of the whole fuel rod throughout its life as well as the localized behaviour of a small part of the fuel rod. FEMAXI-6 consists of two main parts (Fig. 2): one for analysing the temperature distribution, thermally induced deformation, fission product gas release, etc., and the other for analysing mechanical behaviour of the fuel rod. Temperature distribution, radial and axial deformations, fission gas release, and inner gas pressure are calculated as a function of irradiation time and axial position. Stresses and strains in the pellet and cladding are calculated and pellet-cladding mechanical interaction analysis is performed. Moreover, thermal conductivity degradation of the pellet and cladding waterside oxidation are modelled. Its analytical capabilities also cover the boiling transient anticipated in the boiling water reactor (BWR). Elasto-plasticity, creep, thermal expansion, pellet cracking and crack healing, relocation, densification, swelling, hot pressing, heat generation distribution, fission gas release, pellet-cladding mechanical interaction, cladding creep and oxidation are modelled by the code. Users can perform a local pellet-cladding mechanical interaction analysis, such as pellet ridging as an optional process. However, the present analysis conducts the entire fuel rod length analysis [5].

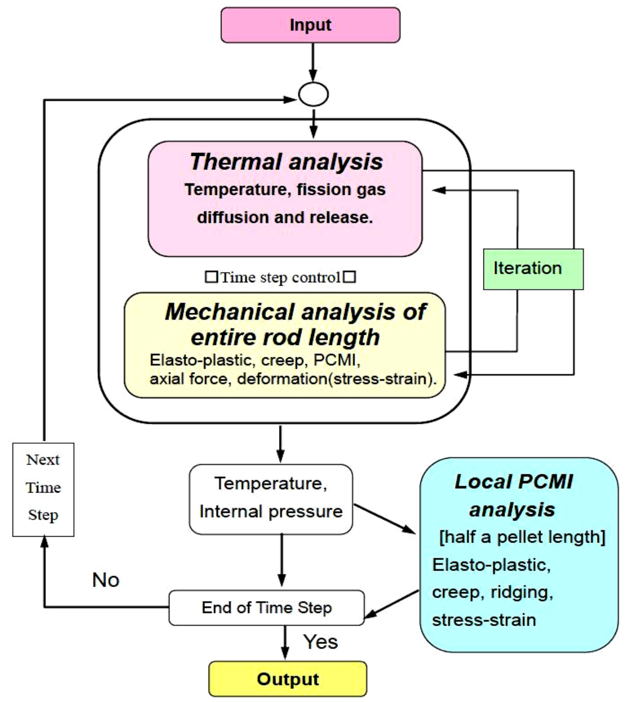

Fig. 2 Entire code structure of FEMAXI-6 [5]

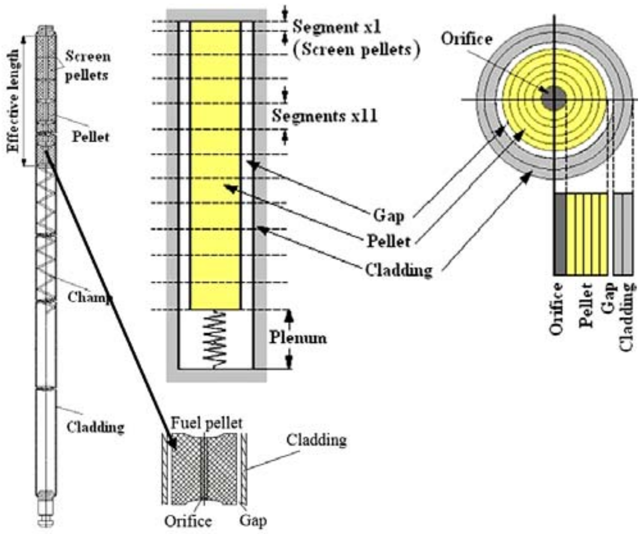

Fig. 3 Model of RBMK-1500 fuel rod (bottom bundle), developed using FEMAXI-6 code [5]

The model of RBMK-1500 fuel rod was developed using FEMAXI-6 code (Fig. 2). For the analysis the fuel rod from the bottom fuel bundle in the average power channel (2.53 MW) was selected. The main parameters of the fuel rod are presented in

Table 1. In the developed fuel rod model, the length is divided into 12 segments and one of them describes the screen pellets (Fig. 3). The top volume of the fuel rod was modelled as a separate segment. This volume contains the clamp, compressing column of the pellets. A more detailed description of the model is presented in [6].

Table 1

Fuel rod parameters

Parameter	Value
Length of the fuel rod, mm	3640
Active length of the fuel rod, mm	3410
Height of the screening pellets, mm	30
Length of the plenum, mm	170
Outside diameter of the fuel rod, mm	13.45
Inside cladding diameter, mm	11.75
Outside fuel pellet diameter, mm	11.5
Pellet central orifice diameter, mm	2
Fuel enrichment in U^{235}, %	2.6
Partition of erbium in fuel, %	0.5
Edge pellet enrichment, %	0.7
Fuel pellet density, g/cm^3	10.55
Mass of fuel within the fuel rod, g	3500
Initial pressure of gases in the fresh fuel rod at cold conditions, MPa	0.5

3. Benchmark of adopted code and developed model

As it has been already mentioned, FEMAXI-6 consists of two main parts used for thermal hydraulic and structural analyses. Because the materials of fuel and cladding and the design of fuel rods in RBMK are different, comparing to the vessel type light water reactors, the FEMAXI-6 code was adopted. The RBMK fuel rod characteristics (thermal conductivity of the fuel pellets and their specific heat dependency on the temperature; thermal conductivity of the fuel rod cladding and its specific heat dependency on the temperature) were included into FEMAXI-6 code. The adaptation of this code was presented in the article [6].

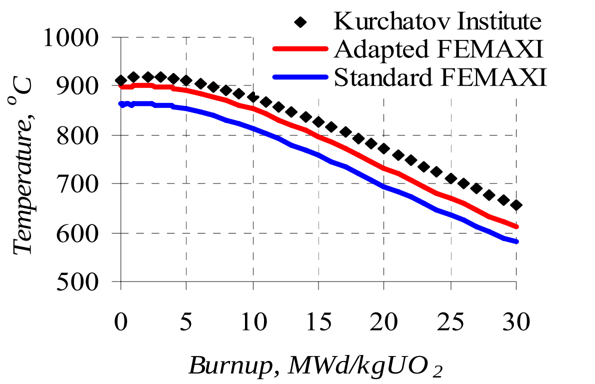

Fig. 4 Temperatures of the fuel centre in the average power channel

Before using the developed fuel rod model for operational transients and accident analysis in Ignalina NPP, the model validation has been performed. The results of fuel parameters dependency on burnup [7], calculated by specialists of Kurchatov Institute, were used as the reference for code to code comparison. More details of this comparison are presented in paper [6].

Different as in paper [6], where the thermal hydraulic part of the code was validated, the validation of FEMAXI-6 code structural analysis of fuel rods is performed in this paper. FEMAXI-6 code can calculate stresses and strains in the pellet and cladding [5]. For the validation of FEMAXI-6 structural analyses part, the received results were compared with the results calculated with BRIGADE/Plus_2.1 code, which uses the finite element method.

BRIGADE is a software package for structural mechanics analysis and design of bridges and civil structures. BRIGADE includes integrated ABAQUS solver technology which guarantees high performance and accuracy [8]. Using one 3D-model to analyze all the load cases saves considerable amount of time in comparison to the traditional 2D analysis tools.

Using BRIGADE code, stresses and displacements were calculated only in the cladding of the fuel rod. For a more simplified model, only one part (height of 0.3 m) of the whole cladding (the part of the fuel rod with the highest energy generation rate) was chosen. This part (segment) is respected by the segment No. 7 (~2 m form the bottom of the active core) in the fuel rod model of FEMAXI-6.

Geometrical data of the cladding (

Table 1), the maximal inner surface temperature, and inner and outer pressure calculated by FEMAXI-6 code were used as the initial conditions to the BRIGADE code model development. According to the FEMAXI-6 code calculation results, the maximal cladding inner surface temperature is 347.1°C, maximal inner pressure – 1.75 MPa and maximal outer pressure – 7.58 MPa. These parameters were reached during 803.8 MWd/tUO_2 burnup in the fuel rod segment No. 7 (~2m form the core bottom).

Table 2

Results of FEMAXI-6 and BRIGADE comparison

	Cladding stress, MPa	Axial displacement, m
FEMAXI-6	40.80	$4.95 * 10^{-5}$
BRIGADE	40.92	$5.5 * 10^{-5}$

Stress and displacements in the cladding calculated with FEMXI-6 and BRIGADE codes were compared (Table 2). The equivalent cladding stress and axial displacement of the fuel rod cladding segment are very similar. The differences occur in the calculation results of radial displacements. The radial displacement calculated using FEMAXI-6 code is two times higher than the results received from the BRIGADE code calculations. It must be noticed that the radial displacement is 10 times smaller comparing to the axial displacement. The differences arise because the BRIGADE code solves only steady state conditions. Cladding creep, oxidation layer and irradiation (parameters which have influence on the radial displacement) are not evaluated in the BRIGADE code. This explains the differences of radial displacement calculated using the FEMAXI-6 and BRIGADE codes.

The comparison of the calculation results, obtained by Lithuanian Energy Institute and Kurchatov Institute (Fig. 4) and with FEMAXI-6 and BRIGADE codes (Table 2) demonstrates that the developed FEMAXI-6 model of RBMK–1500 fuel rod is acceptable for performing structural analysis in Ignalina NPP.

4. Analysis of the processes in RBMK–1500 fuel rod during normal operation

In the RBMK type reactor, fuel assemblies operate for several years until they reach their limit of burnup. During this long–term operation, the reactor power changes several times due to an emergency shutdown or reduction of power. Moreover, the reactor is shutdown once a year for preventive maintenance.

For detailed analysis, the specialists of the Ignalina NPP selected a fuel assembly which has the average power loaded with 2.6% U^{235} enrichment with burnable erbium absorber fuel [9]. Parameters of the assembly were measured at the intervals of about one week. For several typical cases of transient (reactor start–up, the increase/decrease in power, reactor shutdown), the parameters were recorded at the intervals of several minutes. The reactor power history of the second reactor unit of the Ignalina NPP during July 2003-January 2007 is presented in Fig. 5. During this period 50 changes in the reactor power occurred and fuel burnup reached 24000 MWd/tUO_2 in the fuel channel with overage power. Within the time intervals when the reactor was shutdown, the burnup remained approximately constant [9].

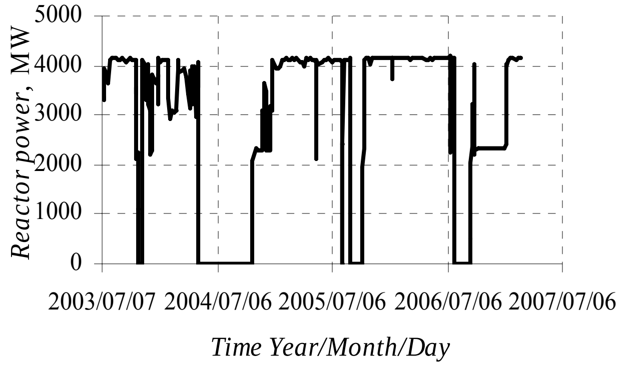

Fig. 5 Power history of the second unit of the Ignalina NPP [9]

Based on the information of reactor power history and the recorded data of operating parameters and burnup (average for a fuel rod), the dependencies of velocity, pressure and temperature of the coolant on burnup were established (Figs. 6-8). Maximum linear load was calculated from reactor power history. As it is shown in Fig. 5, during the whole operation of fuel assembly, the full shutdown of the reactor occurred 5 times. Coolant flow rate through the fuel channel was reduced to 10 m^3/h when the reactor was shutdown. At the shutdown the pressure in the fuel channels is decreasing down to the atmospheric pressure (Fig. 7) and the temperature is reduced to 100°C (Fig. 8).

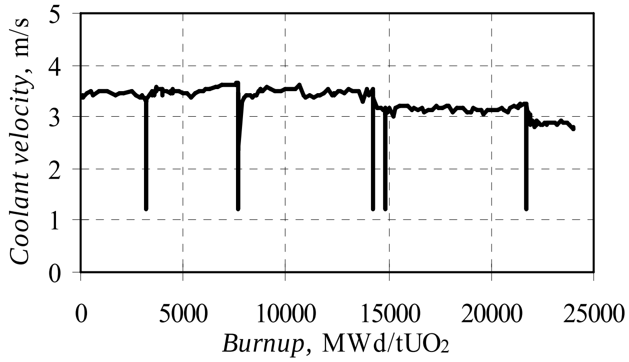

Fig. 6 Coolant velocity history [9]

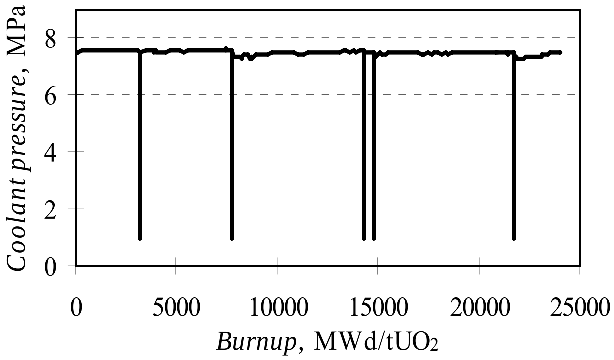

Fig. 7 Coolant pressure history [9]

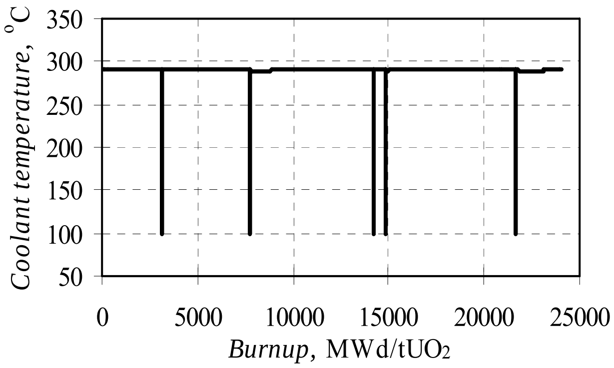

Fig. 8 Coolant temperature [9]

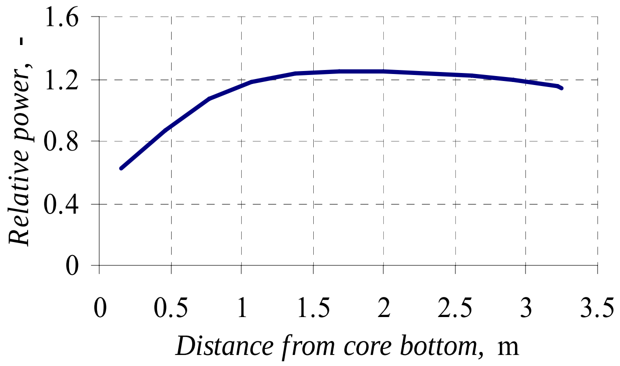

Fig. 9 Axial power profile of the bottom bundle [9]

The analysis was performed for the fuel rod of 2.6% U^{235} enrichment with a burnable erbium absorber from the fuel channel with the average initial power (2.5 MW) from the bottom bundle (

Table 1). The initial fuel rod parameters were assumed as shown in

Table 1; the linear power, coolant velocity, coolant pressure and coolant temperature dependencies on burnup (Figs. 6-8) were used as the initial input data for the FEMAXI–6 calculations. Axial power profile for the fuel bundle is shown in Fig. 9 which presents that the highest energy generation peak is into the middle of the fuel bundle (about 2 m from the core bottom – segment No. 7 in the FEMAXI–6 model).

The behavior of the fuel rod parameters, calculated using the FEMAXI–6 model, is presented in Fig. 10-Fig. 12. In these figures the parameters are presented only in segment No. 7, i.e. the segment with the highest power. The peak temperatures of the fuel rod (Fig. 10) are decreasing due to the decrease of power during the reactor operation (Fig. 5). During the reactor shutdown, the temperatures of the cladding and fuel dropped down to 100°C (coolant saturation temperature at atmospheric pressure): such conditions are preserved during the reactor maintenance.

During the reactor operation, the fuel temperature is decreasing but the pressure of gasses in the gap between the fuel and cladding remains approximately constant (Fig. 11). It happens due to two phenomena: 1) when the gap width between the pellets and cladding decreases (Fig. 13 marked as a reference), the total volume of gases also decreases; 2) increase of the gas pressure because of fission gas release. According to the FEMAXI–6 analysis, the release of fission gases is constant during the whole fuel assembly operation. This means that the amount of gases is increasing. The fraction of gas mixture in the fuel rod is slowly changing. The fraction of He is decreasing from 100% down to 94%, while the fraction of Xe and Kr are increasing from 0 up to 6%.

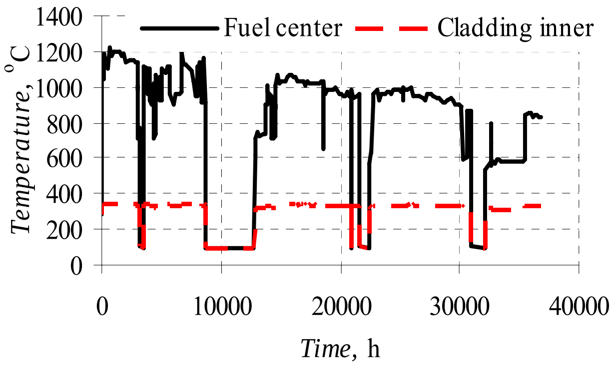

Fig. 10 Peak temperatures in fuel rod (~2 m from the core bottom – segment No. 7 in the FEMAXI–6 model)

The calculations show that the radius of fuel pellets is slightly increasing, while the radius of the cladding is decreasing. Thus, due to radial deformations of the pellets and cladding, the gap between the pellet and cladding is decreasing (Fig. 13 marked as a reference). At the time when the reactor is shutdown, the radial gap is increasing due to the cooldown of the reactor. However, during the reactor operation, the size gap between fuel pellet and the cladding continuously decreasing. This means that fuel pellet expansion and cladding shrinking are not reversible processes. However, it must be noticed that the gap between the fuel pellet and the cladding (segment No. 7) still remains open during the whole normal operation period (Fig. 13 marked as a reference).

The elastic deformation of fuel cladding is very small (Fig. 12). Because during the operation the fuel cladding temperature increases up to ~ 300 °C at the pressure ~ 7 MPa from the side of the coolant, and the pressure of gas inside the fuel rod is ~ 1.7 MPa, the fuel cladding is compressed. Thus, the presented elastic deformations (Fig. 12) are caused by cladding compression (for this reason, the deformations are negative). After the reactor shutdown, the radial dimensions of the cladding return close to the initial conditions. The stresses of the fuel rod cladding are presented in Fig. 14 marked as a reference. The stresses are decreasing in accordance with the decreasing linear power. As it is stated in [10], the yield stress for Zr+1 % Nb alloy is 180 – 220 MPa for 300°C temperature and 320-380 MPa for 20°C temperature. After exceeding this yield stress limit, the fuel cladding will be affected by the plastic deformation that leads to cladding failure. In the case under examination, the calculated maximal value of equivalent stress in the cladding is much lower than the yield stress. Thus, the fuel cladding remains intact in the whole life of the fuel assembly.

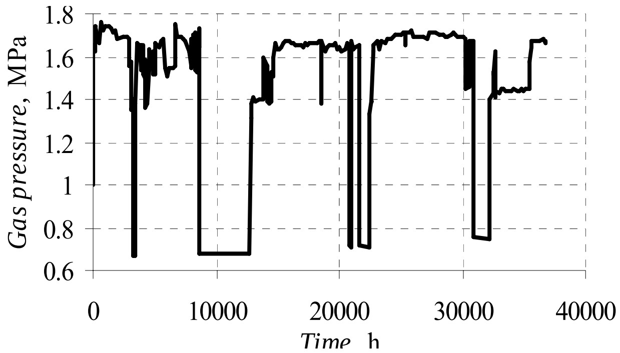

Fig. 11 Gas pressure in the fuel rod

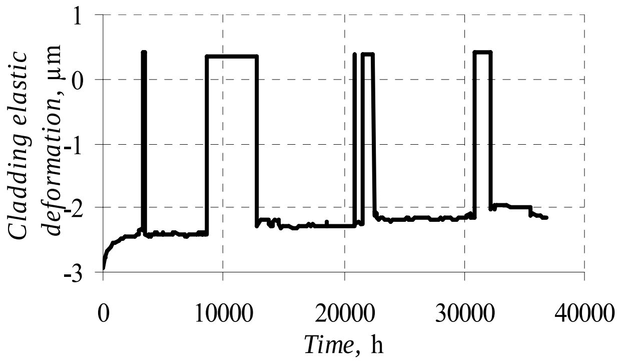

Fig. 12 Elastic deformation of the cladding in radial direction

5. Uncertainty and sensitivity analysis

Any calculation results approximate the real physical behaviour only with a limited accuracy. This is due to the model deficiencies and model simulations or the uncertainties related to the input data of parameter values, which are not known exactly. In order to evaluate the uncertainties of calculation, statistical methods are often used. The GRS methodology [11] and computer code of statistical methods SUSA 3.5 [12] were used for the sensitivity and uncertainty analysis of the FEMAXI-6 calculation results. This method also allows to evaluate the impact of the uncertainties of input data on final results.

Input parameters which can have an impact on the calculation results, range, deviation and probability distribution are shown in Table 3. In this table the ranges of parameters, namely: inside cladding, fuel pellet and fuel pellet central orifice diameters, initial pressure of gases in the cladding, cladding thickness and theoretical density ratio are grounded on the source [13. It was assumed that the power of the fuel rod, and mass flow and pressure of the coolant can wary in the intervals ±3%, ±2% and ±1% respectively. Such range of the mentioned input parameters was determined according to the experience of safety analysis calculations of the Ignalina NPP [9].

For all parameters normal probability distribution is selected. The mean deviation of the normal distribution is determined according to the following formula

$$s = (a - i)/6 \qquad (1)$$

where s is mean deviation, a is maximal value of the parameter, i is minimal value of the parameter.

Table 3

List of the uncertain input parameters

#	Input parameter	Range		Deviation (s)	Distribution
		Minimal value (i)	Maximal value (a)		
1	Inside diameter of the cladding, mm	11.7	11.8	0.1	Normal
2	Initial pressure of gases in the cladding, MPa	0.47	0.7	0.04	Normal
3	Fuel pellet diameter, mm	11.44	11.52	0.01	Normal
4	Central orifice diameter of the pellet, mm	1.9	2.3	0.07	Normal
5	Theoretical density ratio	0.95	0.98	0.01	Normal
6	Power profile, %	-3	3	1	Normal
7	Cooling water temp., %	-1	1	0.33	Normal
8	Cooling water pressure, %	-1	1	0.33	Normal
9	Cooling water velocity, %	-2	2	0.67	Normal
10	Cladding thickness, mm	1.65	1.93	0.3	Normal

It was assumed that the parameters which may impact the calculation uncertainty are independent.

Using SUSA program code [12], 60 collections of input parameters were composed. The number of runs necessary for one-side or two-side tolerance intervals depends only on the required probability and confidence level of the statistical tolerance limits. The relationship between these parameters is described by Wilks formula [14]. In the case under examination, the uncertainty analysis was performed using a one-side tolerance limit (with 0.95 of probability and 0.95 of confidences). For each collection an input file was composed for the FEMAXI-6 code and the performed calculation. The uncertainty analysis was performed to the following calculation results: the deviation of cladding stress and decrease of the gap between the pellet and cladding (Figs. 13-14). These figures present the maximum appropriate one side higher tolerance limit, minimum - one side lower tolerance limit with 0.95 of probability and 0.95 confidences and reference means taken from the previous calculation which was presented in paragraph 5. The calculated minimal value of the gap between the fuel pellet and cladding reaches 0 at the end of the calculations (Fig. 13). This means that at the end of the operation of the analyzed case (fuel assembly from the average power load channel containing 2.6% U^{235} enrichment with the burnable erbium absorber fuel [9]), an interaction between the pellet and cladding could occur. However, the calculated maximal value of the equivalent stress in the cladding is still much lower than the yield stress (Fig. 14). As it is shown in the previous chapter, the yield stress for Zr + 1 % Nb alloy is 180 – 220 MPa for 300 °C temperature [10]. Thus, the cladding is not affected by the plastic deformation and remains intact in the whole life of the fuel assembly.

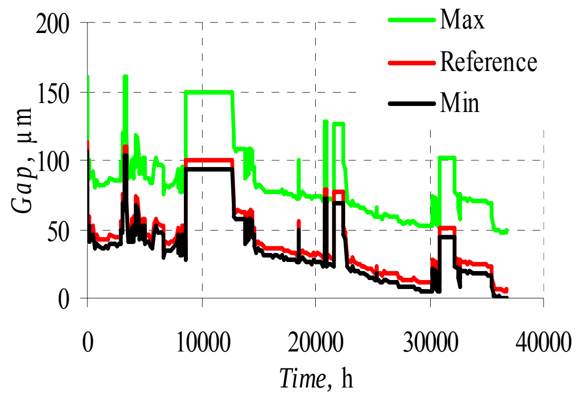

Fig. 13 Minimum, maximum and reference value of the gap between the fuel pellet and cladding

Based on the results of the performed calculations, the impact of input parameters on the calculation results was analysed, applying Spearman's rank correlation coefficient method. This method shows the quantity of the impact made by the input parameters on the calculation results. A negative and positive parameter impact on the results could be observed. The positive impact means that when the parameter value is increasing, the result value also rises, whereas in the case of the negative impact, the increase of the parameter value leads to the decrease of the result value. Coefficient of determination (R^2) with respect to Spearman's rank correlation shows the independence of the parameters on each other. In practice, it is often required that the linear model determination ratio should be at least 0.6. If R^2 is less, then the standardized regression coefficient of the sensitivity ranking of the parameters may be incorrect.

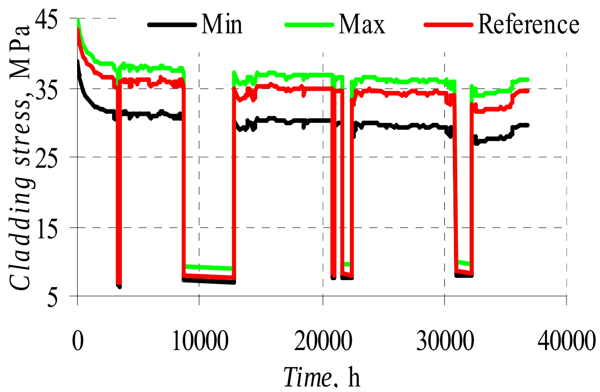

Fig. 14 Minimum, maximum and reference value of the equivalent stress of the cladding

In the presented analysis, the coefficient of determination (R^2) with respect to Spearman's rank correlation is more than 0.9 in both the cladding stress and the gap between the pellet and cladding. Thus, the standardized regression coefficient of the sensitivity ranking of the parameters can be applied. The important parameters to the gap between the fuel pellet and cladding and the equivalent stress are presented in Figs. 15, 16. These figures present only the parameters which have the highest impact.

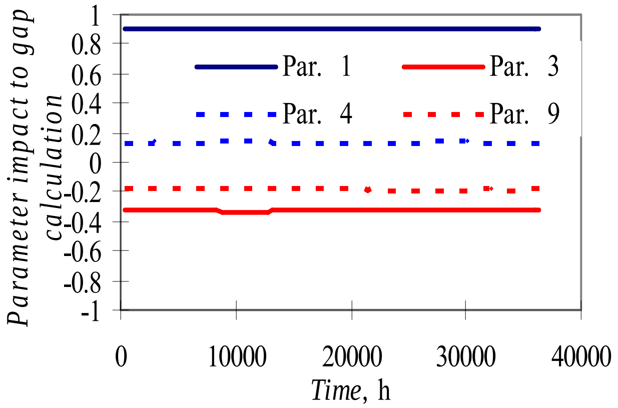

Fig. 15 Impact of input parameters to calculated gap between fuel pellet and the cladding. Parameter numbers according Table 3

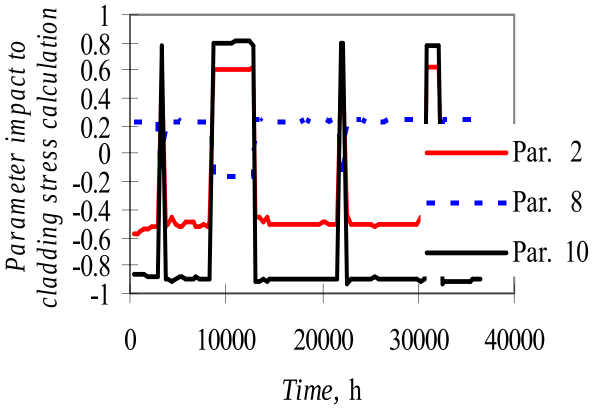

Fig. 16 Impact of the input parameters to the calculated equivalent stress of the cladding. Parameter numbers given according to Table 3

According to Spearman's rank correlation coefficient (Fig. 15), the following parameters: inside cladding diameter *1*, fuel pellet diameter *3*, pellet central orifice diameter *4* and cooling water velocity *9* have the greatest impact on the gap between the pellet and cladding. Other parameters have insignificant impact. Geometrical data of the fuel rod have the greatest impact on the gap calculation and these parameters should be chosen more precisely. Cladding thickness *10,* initial pressure of gases in the cladding *2* and cooling water pressure *8* has the greatest impact on the cladding stress (Fig. 16). As it is shown in Fig. 16, in the case of normal operation and reactor shutdown stage, the impact of the parameters changes its characters. During the normal reactor operation, cladding thickness and initial pressure of gases in the cladding have a negative impact, while cooling water pressure has a positive impact on the cladding stress calculations, but during the reactor shutdown stage, these parameters have an opposite impact. This is because during the operation and shutdown stages, the nature of cladding stresses are different:

- during normal operation the pressure outside the cladding is higher and difference between outer and inner side is ~5.8 MPa, thus the cladding is affected by "compression";

- at the shutdown stage pressure the pressure inside fuel rod is higher as outside – the pressure, difference (from the inner and outer side of the cladding) is ~0.6 MPa and the cladding is affected by “ballooning”.

6. Summary and conclusions

The developed fuel rod model for the Ignalina NPP and FEMAXI-6 code, with included thermal properties of the fuel pellets and cladding, used in RBMK-1500 reactor, were applied for thermal hydraulic and structure analyses of the fuel rod. These two possibilities of the FEMAXI-6 code were validated in the following manner. The thermal hydraulic part was validated using the comparison of the FEMAXI-6 calculation results and Kurchatov Institute calculation results, while for the structural part the FEMAXI-6 calculation results were compared to the received calculation from the BRIGADE code. The FEMAXI-6 calculation results are in a good agreement with the received data from Kurchatov Institute and BRIGADE code. This leads to the conclusion that the adapted FEMAXI-6 code version and the developed model are suitable for the analysis of the operational transients and accidents in the fuel rods of RBMK-1500.

The analysis of the processes in RBMK-1500 fuel rod during the whole life of the fuel assembly was performed for the fuel rod of 2.6% U^{235} enrichment with burnable erbium absorber from the fuel assembly with the average initial power (2.5 MW) from the bottom fuel bundle. The best estimate calculation results, using GRS methodology and program package SUSA of fuel rod, during the whole life of normal operation showed that:

- in the reference calculation, the gap between the fuel pellet and cladding remains open during the whole normal operation period;
- the performed uncertainty analysis shows the possible gap closure at the very end of the fuel assembly operation in the reactor core. However in the “worst” cases (lower tolerance limit with 95% of probability and 95% of confidences) the stress in the cladding is not exceeding 45 MPa, while the yield stress of Zr+1%Nb alloy is 180-220 MPa for 300°C temperature – this is significant higher.

The results of the analyses lead to the conclusion that the safety barrier is sustained: the fuel cladding remains intact during the whole life in the normal operation.

The impact of the input parameters on the results of the calculations has shown that:

- the inside cladding and fuel pellet diameters have the greatest impact on the gap between the pellet and cladding;
- cladding thickness, initial pressure of gases in the cladding and cooling water pressure have the greatest impact on the cladding stress;
- the impact coefficients of the parameters change their characters in the case of normal operation and reactor shutdown stage. This is because the nature of the cladding stress is different during the operation and shutdown stages.

In future, the results obtained from the analysis of the normal operation of the fuel rod could be used as the initial conditions for the simulation of the processes in the fuel rods stored in the spent fuel pools.

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GERIAUSIO ĮVERČIO METODOLOGIJOS TAIKYMAS TIRIANT EKSPLOATACIJOS METU RBMK REAKTORIŲ ŠILUMĄ IŠSKIRIANČIUOSE ELEMENTUOSE VYKSTANČIUS PROCESUS

R e z i u m ė

Straipsnyje pristatoma normalios eksploatacijos metu RBMK reaktorių šilumą išskiriančiuose elementuose vykstančių procesų analizė, atlikta taikant geriausio įverčio metodologiją. Šiai analizei atlikti pasirinktas programų paketas FEMAXI-6. Pirmiausia, atsižvelgiant į RBMK reaktorių ŠIEL‘ų specifiką, programų paketas buvo pritaikytas. Vėliau, sudarytas RBMK ŠIEL‘o modelis ir, naudo-

jantis pritaikytu programų paketu FEMAXI-6, ištirti procesai vykstantys ŠIEL‘e kuro rinklės, eksploatacijos metu. Šiai analizei pasirinktas ŠIEL‘as iš vidutinės galios (2,5 MW) kuro rinklės.

Apvalkalo įtempių ir tarpelio tarp kuro tablečių ir apvalkalo elgsenos jautrumo ir neapibrėžtumo analizė atlikta taikant GRS geriausio įverčio metodologiją ir programų paketą SUSA. Analizės rezultatai parodė, kad ŠIEL‘o apvalkalas, įvertinant galimas skaičiavimo neapibrėžtis, normalios eksploatacijos metu išlieka nepažeistas. Atlikta analizė rodo, kad galima nustatyti ŠIEL‘ų būklę po normalios eksploatacijos, kas yra būtina žinoti norint kurą ilgą laiką saugoti panaudoto kuro baseinuose.

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BEST ESTIMATE ANALYSIS OF PROCESSES IN RBMK FUEL RODS DURING OPERATION CYCLE

S u m m a r y

In this article the processes in fuel rods of RBMK-type reactors during operation cycle were analyzed employing the best estimate methodology. The FEMAXI–6 code was selected for such analysis. At first, the evaluation of the specifics of RBMK fuel rods and the adaptation of the code was provided. Later, the single fuel rod model of RBMK-1500 was developed and the processes, which occur during whole life of fuel assembly inside the core, were analyzed, using adopted FEMAXI-6 code. For this analysis the fuel rod from fuel assembly with average initial power (2.5 MW) was selected.

The uncertainty and sensitivity analysis of the behavior of cladding stress and gap between fuel pellet and the cladding were performed using GRS best estimate methodology by employing SUSA program code. The results of the analysis show, that the fuel cladding remains intact during whole life in normal operation, evaluating possible uncertainties of the calculation. The performed analysis demonstrates a possibility to identify the state of fuel rods after the normal operation that is necessary before the long-term fuel storage in the spent fuel pools.

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