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Analysis of processes in RBMK-1500 fuel assemblies during normal operation cycle

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Before transferring fuel rods from the reactor core to the spent fuel pool, the state of the fuel rods after operation should be determined. The state of a fuel rod could be estimated by employing deterministic analysis. In this work, the state of the RBMK-1500 fuel rod after normal operation was evaluated using the numerical method.

Recently, analysis of the processes in RBMK-1500 fuel rods has been performed only by RBMK designers. In the last years, specialists of the Lithuanian Energy Institute started modelling the process in fuel rods of the Ignalina Nuclear Power Plant (NPP) as well, using the FEMAXI-6 code adapted to the specific features of the RBMK-1500 fuel rod. The calculation results obtained using the adapted FEMAXI-6 code were compared with calculations performed by specialists from the Kurchatov Institute. A comparison of the results shows that the adapted FEMAXI-6 code is suitable for the analysis of processes in fuel rods of the RBMK-1500.

The processes during the whole life of fuel rods were analyzed using the adapted FEMAXI-6 code. For this analysis, a fuel rod from the fuel channel with an average initial power (2.5 MW) was selected. The state of a fuel rod after operation was evaluated using numerical analysis. The obtained results can be used for the analysis of processes in fuel rods stored in spent fuel pools.

Key words: RBMK-1500, fuel rod, FEMAXI-6

1. INTRODUCTION

The Ignalina NPP is the only nuclear power plant in Lithuania. It consists of two units commissioned in 1983 and 1987. The Ignalina NPP Unit 1 was shut down for decommissioning at the end of 2004, and Unit 2 operated until the end of 2009. Both units were equipped with channel-type graphite-moderated boiling water reactors RBMK-1500. According to the design, the spent fuel should be returned for reprocessing to Russia. However, no fuel assembly has been taken away from the territory of the Ignalina NPP, and all assemblies of spent fuel are stored in the spent fuel pools and dry storage facility on-site.

Each reactor unit is equipped with a system of spent fuel pools. To ensure the safe storage of the facility, it is necessary to determine the real state of fuel rods (integrity, etc.) after normal operation before transferring them from the reactor core to spent fuel pools. The condition of a fuel rod can be estimated by employing deterministic analysis.

Analysis of the processes in RBMK-1500 fuel rods has been performed only by Russian specialists from the Research and Development Institute of Power Energy (general designer of the RBMK) and the Kurchatov Institute using Russian in-house codes. Specialists of the Lithuanian Energy Institute started to model processes in fuel rods of the Ignalina NPP as well. The FEMAXI-6 code is used for the modelling. This code is designed for vessel-type reactors. The materials of fuel and cladding and the design of fuel rods in RBMK-1500 are different compared to vessel-type light water reactors.

The fuel channels in RBMK-1500 are placed in a graphite stack which consists of 2488 graphite columns with vertical bore openings. These openings are used for positioning the channels, which in turn are used for placing fuel assemblies, control rods and several types of instruments into the core [1]. In the beginning, the Ignalina NPP was operated with 2.0% U^{235} enrichment fuel. Later, at the Ignalina NPP the fuel of 2.4%, 2.6% and 2.8% U^{235} enrichment with a burnable erbium absorber was used. The nuclear fuel was compressed into pellets 11.46 mm in diameter and 15 mm in height [1]. The shape of a pellet is adapted to an intensive, high-temperature operating mode. A 2 mm hole through the axis of the pellet reduces the temperature in the centre of the pellet. The pellets were placed into a cladding with the outside diameter 13.6 mm, wall thickness 0.9 mm and active length 3.41 m (Fig. 1). The fuel cladding was made of a zirconium and niobium alloy (Zr + 1% Nb). In the RBMK reactor, the fuel assembly is fit into a circular fuel channel with the internal diameter 80 mm and the core height 7 m. In order to achieve the required height, the RBMK fuel assembly consisted of two fuel bundles placed one above another (Fig. 1). Each fuel bundle included 18 fuel rods placed in two circles around the carrying rod.

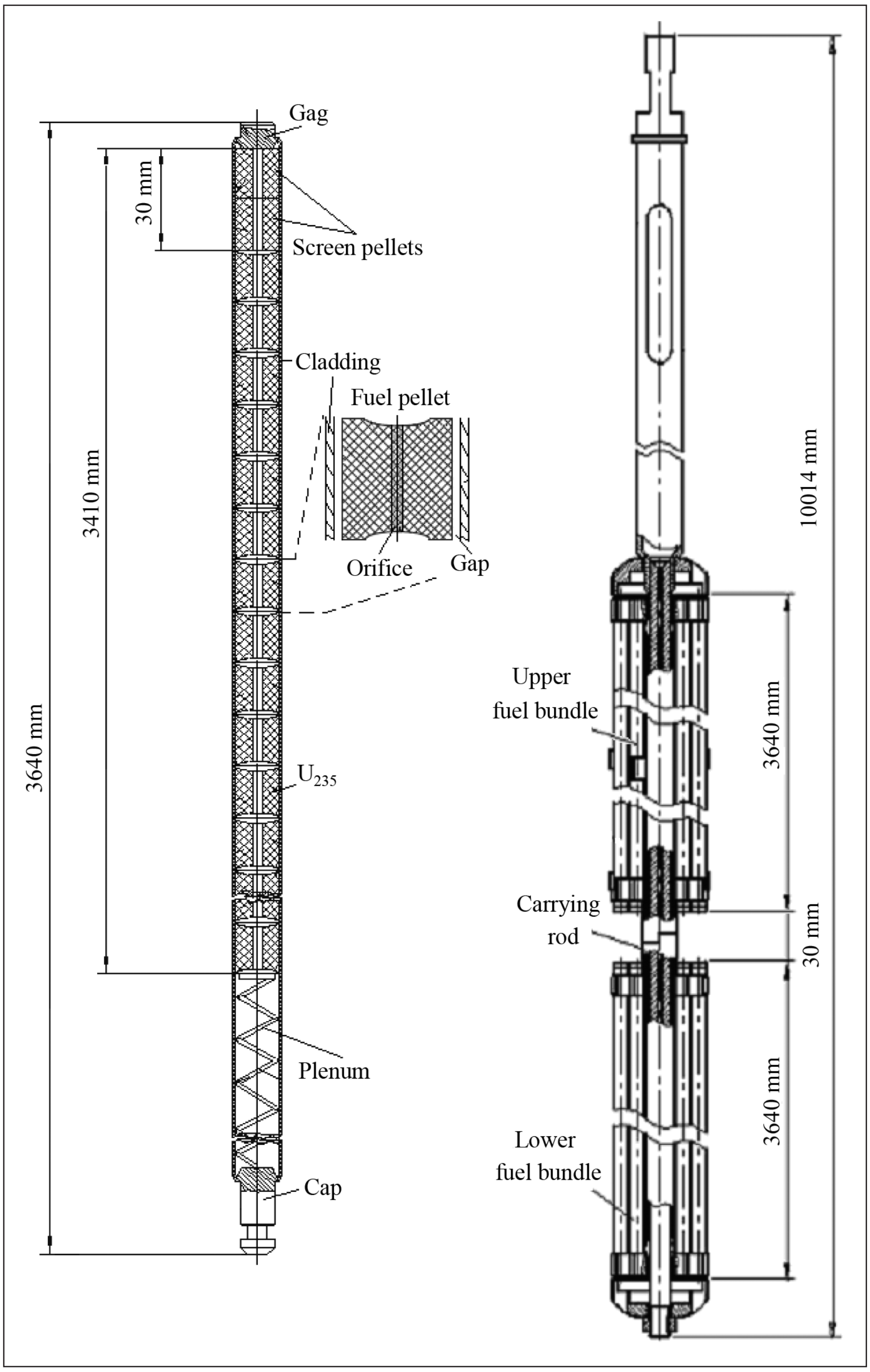

Fig. 1. Fuel rod and fuel assembly of RBMK-1500 reactor [2]

2. THE FEMAXI-6 CODE

In the world, a several codes are used for modelling the processes in fuel rods, e. g. FEMAXI, TESPA-ROD, FUELSIM, TRANSURANUS, etc. These codes were widely used for the analysis of the thermal and mechanical processes that occurred in fuel rods during operation and transients. Several codes (TESPA-ROD, FUELSIM and FEMAXI) were tested and the results were compared [4]. The FEMAXI code provided the most accurate results. In addition, this code can simulate a wider range of fuel rod events, including normal operation and accidents. The algorithm of the FEMAXI code is quite simple and can be edited or supplemented. Thus, the FEMAXI-6 code can be adapted to simulate processes in the RBMK-1500 fuel rod. For these reasons, FEMAXI-6 code was chosen for the further simulation.

The FEMAXI-6 code can analyse the integral behaviour of a whole fuel rod throughout its life as well as the localized behaviour of a small part of a fuel rod [5]. The FEMAXI-6 consists of two main parts (Fig. 2): one for the analysis of temperature distribution, thermally induced deformation, and a fission product gas release, etc., and the other for analysing the mechanical behaviour of a fuel rod. Temperature distribution, radial and axial deformations, fission gas release and inner gas pressure were calculated as a function of irradiation time and axial position. Stresses and strains in the pellet and cladding were calculated, and the pellet–cladding mechanical interaction analysis was performed. Also, the thermal conductivity degradation of the pellet and cladding waterside oxidation were modelled. Elastoplasticity, 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 can be modelled by the FEMAXI-6 code.

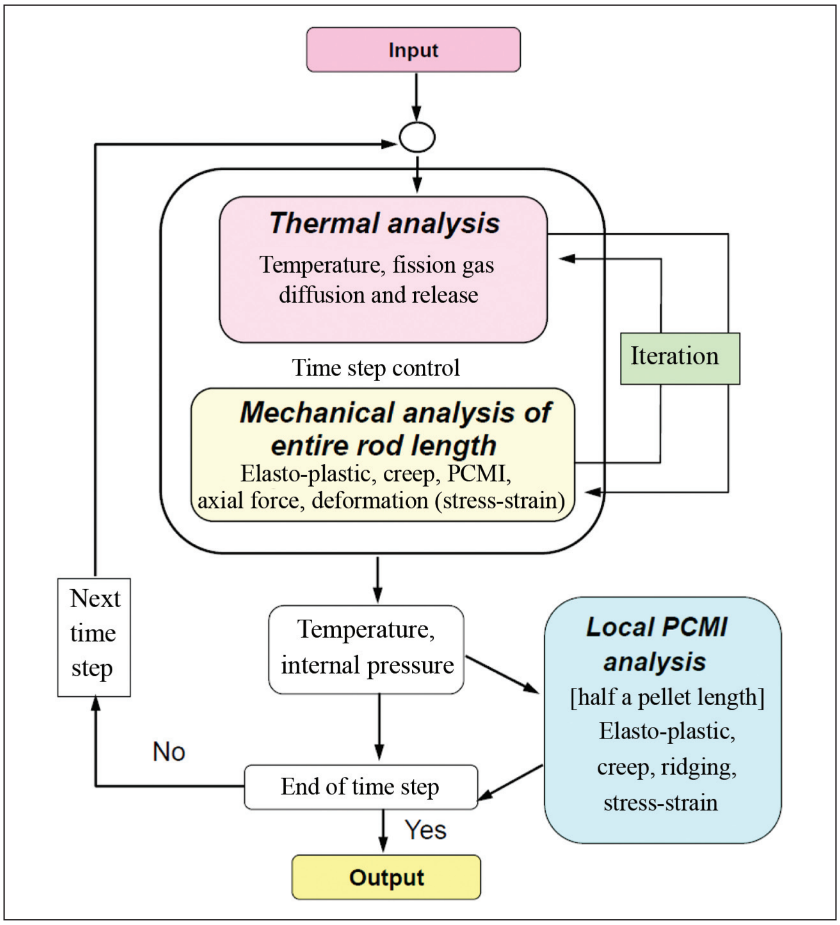

Fig. 2. Entire code structure of FEMAXI-6 [5]

3. A MODEL OF THE RBMK-1500 FUEL ROD

As already mentioned, the FEMAXI-6 code is designed for BWR type reactors. The materials of fuel and cladding and the design of fuel rods in the RBMK-1500 differ from those of vessel-type LWR, therefore, the FEMAXI-6 code was adopted by incorporating the thermal properties of fuel rod pellets and cladding [6].

The model of an RBMK-1500 fuel rod was developed using the FEMAXI-6 code. For the analysis, a fuel rod from the bottom fuel bundle in the average power channel (2.5 MW) was selected.

To assess the influence of the model nodalisation several calculations were performed. Fuel rod was divided into a different number of segments (3–40). One of the segments describes two screening pellets (size of segment 30 mm). For all model nodalisations, the segment with screen pellets was the same. The remaining length of a fuel rod was divided into a number of segments different size (Table 1). All the other initial conditions were the same.

The temperature in the pellet centre for each segment was calculated and averaged (Fig. 3). The biggest difference was observed when the fuel rod was divided into 3 and 5 segments. The other calculation variants gave very similar results with a difference of ~0.3%. It can be concluded that a fuel rod should be divided into 10 (11% of the active length of a fuel rod) or more segments.

Considering the previous investigations, the developed fuel rod model in axial direction was divided into 12 segments, one of which describes screen pellets (Fig. 4). The FEMAXI-6 code gives the possibility to divide a fuel rod into segments of different length, so if for a definite point (length)

Table 1. **Adaptation of fuel rods to the proposed model**

Number of segments (including segment with screen pellets)		3	5	10	15	20	25	30	35	40
Height of segment (Segment-n)	mm	1705	852.5	378.9	243.6	179.5	142.1	117.6	100.3	87.4
	% of active length	50	25	11	7	5	4	3.4	3	2.6
Height of screen pellet segment (Segment-1)		30 mm (1% of active length)								

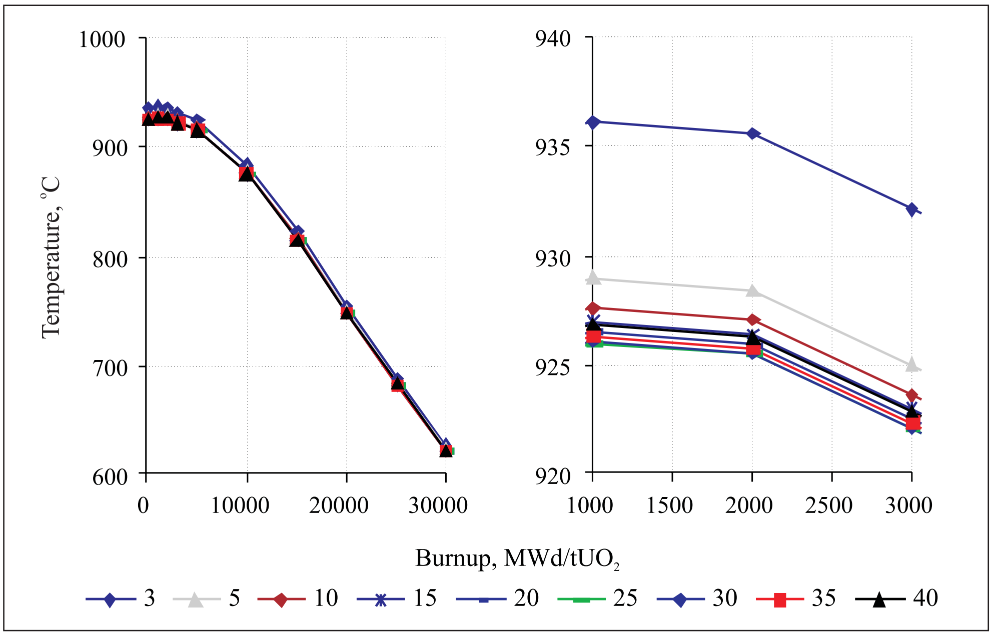

Fig. 3. Average temperature of fuel centre

a more accurate analysis is needed, it can be done by modelling this part with a bigger number of segments of a smaller length. The top volume of fuel rod was modelled as a separate part. This volume contains a clamp compressing a column of pellets. The parameters of fuel pellets and fuel rod cladding are presented in detail in Table 2.

The FEMAXI-6 code has an internal structure in which the finite element mechanical analysis of the entire fuel rod length and thermal analysis are coupled using iteration in each time step for prediction of fuel behaviour, particularly in the high burnup region, i. e. the temperature and fission gas calculation use the gap size and contact pressure obtained by the mechanical analysis of the entire fuel rod length. The FEMAXI-6 code can perform a local pellet–cladding mechanical interaction analysis, such as pellet ridging as an optional process. However, in the present study, only the entire fuel rod length was analysed.

4. VALIDATION OF ADOPTED CODE AND DEVELOPED MODEL

As already mentioned, until recently analysis of the processes in RBMK-1500 fuel rods has been performed only by the Research and Development Institute of Power Engineering (RDIPE) and the Kurchatov Institute specialists. With the FEMAXI-6 code, the LEI starts performing such kind of analysis as well. The adaptation of this code (inclusion of thermal material properties of RBMK-1500 fuel pellets and cladding into the code) was presented elsewhere [6]. Before using the developed fuel rod model for operational transients and accident analysis at the Ignalina NPP, the model validation had been performed. The data on fuel parameters' dependence on burnup [7], calculated by specialists of Kurchatov Institute, were used as a reference for code-to-code comparison. The analysis was performed for a fuel rod from a fuel channel with the average initial power (2.53 MW).

The parameters of fuel rods, linear fuel rod power dependence on burnup and the axial power profile for a bottom fuel bundle modelled using the FEMAXI-6 code were assumed to be the same as in Kurchatov Institute calculations [7]. One can see in Fig. 5 that the energy generation peak is shifted to the very top of the fuel bundle.

Using the FEMAXI-6 code, the centre temperature of a fuel pellet was calculated in all segments, and its average level was fund to be similar to the data obtained at the Kurchatov Institute (Fig. 6), with the largest difference not exceeding 7%. The gap between a fuel pellet and cladding increased in

Table 2. **Fuel rod parameters**

Parameter	Value
Length of fuel rod	3640 mm
Active length of fuel rod	3410 mm
Height of screening pellets	30 mm
Length of plenum	170 mm
External diameter of fuel rod	13.55 mm
Internal cladding diameter	11.75 mm
External fuel pellet diameter	11.5 mm
Pellet central orifice diameter	2 mm
Fuel enrichment in U^{235}	2.6%
Content of erbium in fuel	0.5%
Edge pellet enrichment	0.7%
Fuel pellet density	10.55 g/cm^3
Mass of fuel within a fuel rod	3500 g
Initial pressure of gases in the fresh fuel rod in cold conditions	0.5 MPa

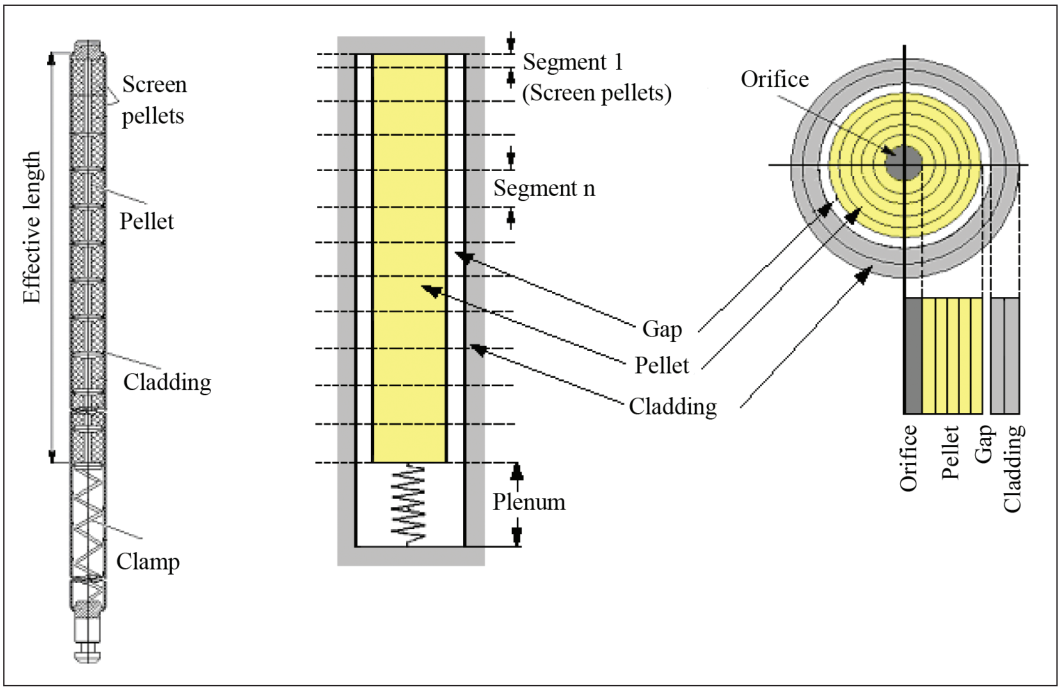

Fig. 4. Model of RBMK-1500 fuel rod (bottom bundle) developed using FEMAXI-6 code

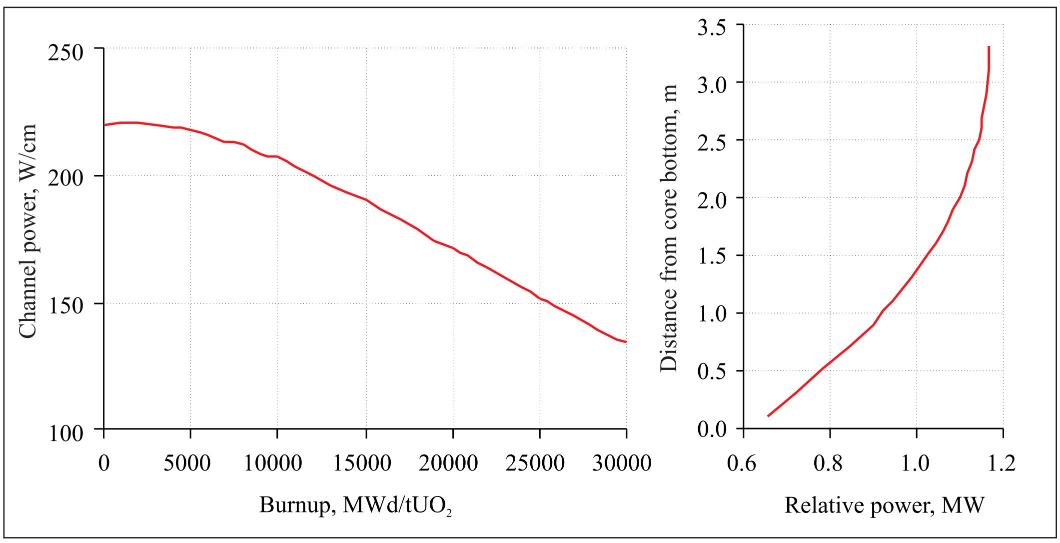

Fig. 5. Relative power and axial power profile in average power channel [7]

the low burnup region (below 7 MWd/kgU) because of the fuel pellet densification phenomenon (Fig. 7). Later, the gap decreased with the burnup. The behaviour of gas composition, calculated using the FEMAXI-6 code, was similar as in the Kurchatov Institute calculation (see Fig. 8).

A comparison of calculation results obtained by the Lithuanian Energy Institute and Kurchatov Institute demonstrated a reasonable agreement: the developed model provided an acceptable prediction as all major trends and phenomena had been predicted correctly. Thus, according to the Adequacy Standard presented in the guidelines for performing code validation and issued by Department of Energy International Nuclear Safety Centre [8], the proposed FEMAXI-6 model of RBMK-1500 fuel rod is acceptable for performing an analysis at the Ignalina NPP.

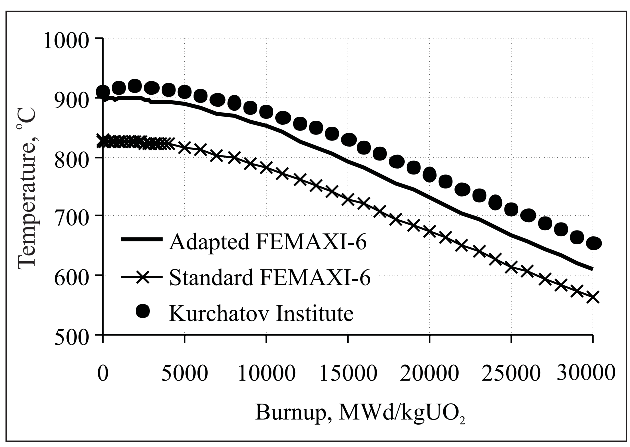

Fig. 6. Fuel centre temperatures in average power channel

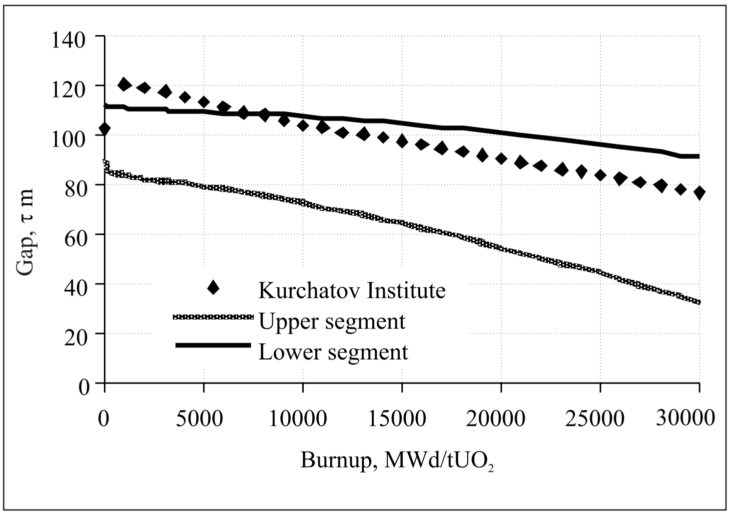

Fig. 7. Gap between fuel pellet and cladding in average power channel

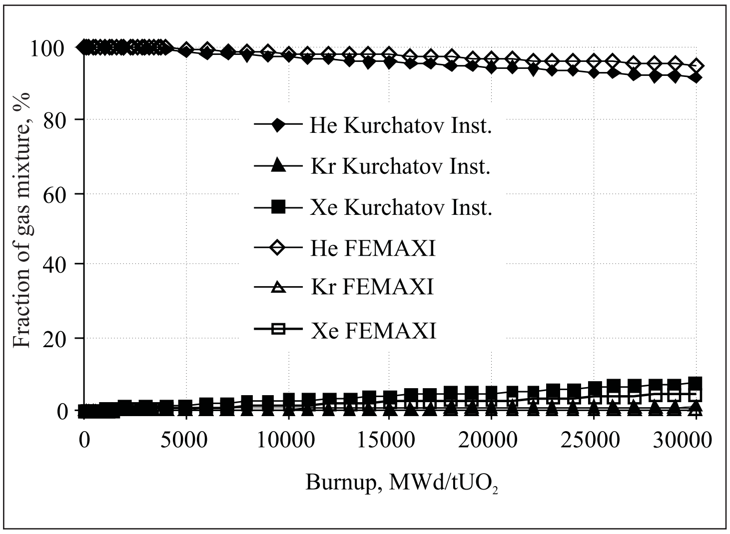

Fig. 8. Gas inside fuel rod in average power channel

5. ANALYSIS OF PROCESSES IN RBMK-1500 FUEL ROD DURING NORMAL OPERATION

In RBMK type reactors, fuel assemblies operate for several years until reaching their limit of burnup. During this long-term operation, the reactor power changes several times because of an emergency shutdown or power reduction. Also, the reactor is shut down once a year for preventive maintenance.

For a detailed analysis, specialists of the Ignalina NPP selected fuel assembly with average an power loaded with 2.6% U^{235} enrichment with burnable erbium absorber fuel [9]. The parameters of the assembly were measured at intervals of about one week. For several typical cases of transient (reactor start-up, an increase / decrease in power, reactor shutdown), the parameters were recorded at intervals of several minutes. The reactor power history of the second unit of the Ignalina NPP during July 2003 – January 2007 is presented in Fig. 9. During this period, 50 changes in reactor power occurred, and fuel burnup reached 24 000 MWd/tUO_2 in a fuel channel with an average power. In the time intervals when the reactor was shut down, the burnup remained approximately constant [9].

Based on the information on the reactor power history and recorded data on the operating parameters and burnup, dependencies of the maximum linear load and coolant velocity, were established (Figs. 10, 11). As is shown in Fig. 10, during the whole operation of the fuel assembly, a full shutdown of the reactor occurs five times – at the burnup of 4 000, 8 000, 14 000, 15 000 and 22 000 MWd/tUO_2. The coolant flow rate through

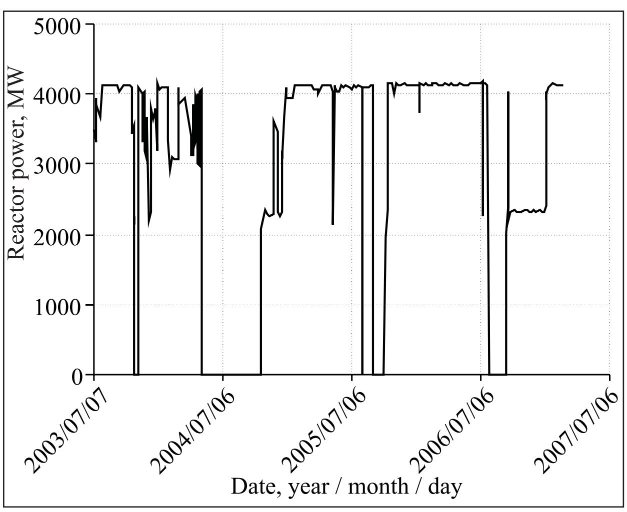

Fig. 9. Power history of the second unit of Ignalina NPP [9]

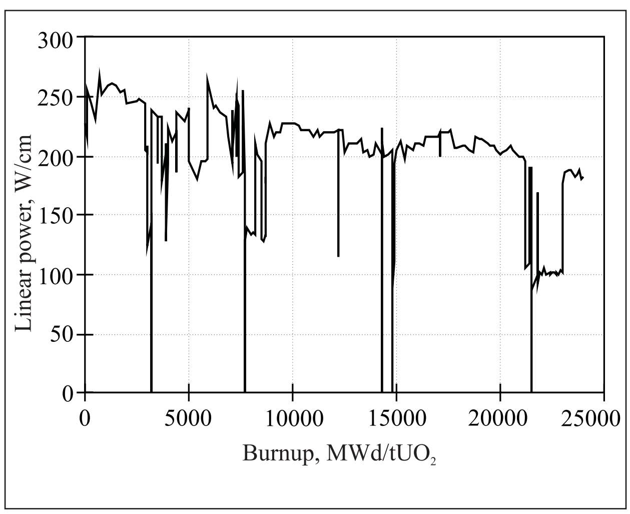

Fig. 10. Linear power history [9]

a fuel channel was reduced to 10 m^3/h when the reactor had been shut down. At shutdown, the pressure in fuel channels decreases to the atmospheric level and the temperature to 100 °C.

The analysis was performed for a fuel rod of 2.6% U^{235} enrichment with a burnable erbium absorber from a fuel channel with the average initial power (2.5 MW) from the bottom bundle. The initial fuel rod parameters were assumed as shown in Table 2; the linear power and coolant velocity dependence on burnup (Figs. 10, 11) were used as initial input data for FEMAXI-6 calculations. During the reactor operation coolant pressure was approximately 7.5 MPa and coolant temperature ~290 °C. The axial power profile for a fuel bundle is presented in Fig. 12 which shows that the energy generation peak is in the middle of the fuel bundle (about 2 m from core bottom – segment No. 7 in the FEMAXI-6 model).

The behaviour of fuel rod parameters calculated using the FEMAXI-6 model is presented in Figs. 13–21. In these figures, the parameters are presented only for segment No. 7, i. e. a segment with the highest power. The peak temperatures of the fuel rod (Fig. 13) decreased due to a decrease of power during reactor operation (Fig. 10). During the periods when the reactor was shut down, the temperatures of cladding and fuel decreased down to 100 °C (coolant saturation temperature at atmospheric pressure); such conditions were kept during reactor maintenance (Fig. 13).

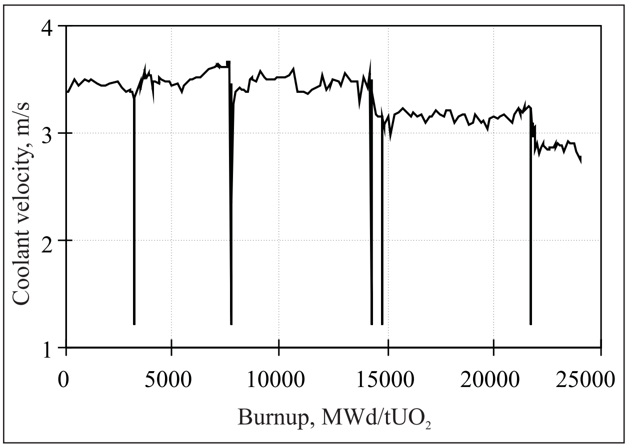

Fig. 11. Coolant velocity history [9]

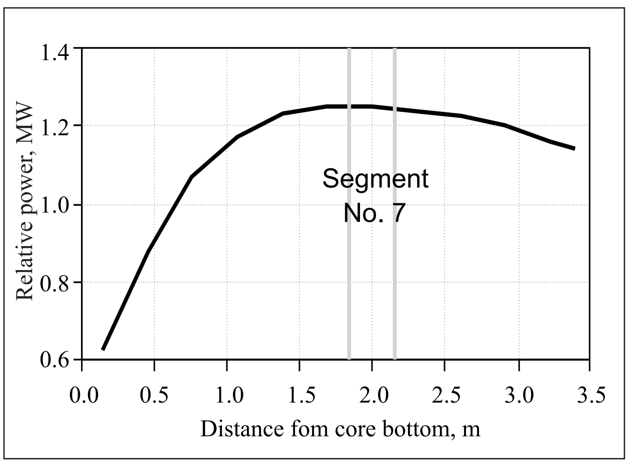

Fig. 12. Axial power profile of the bottom bundle [9]

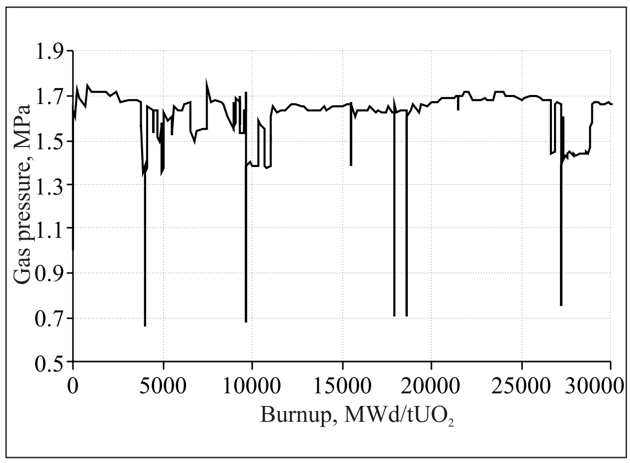

Fig. 14. Gas pressure in fuel rod

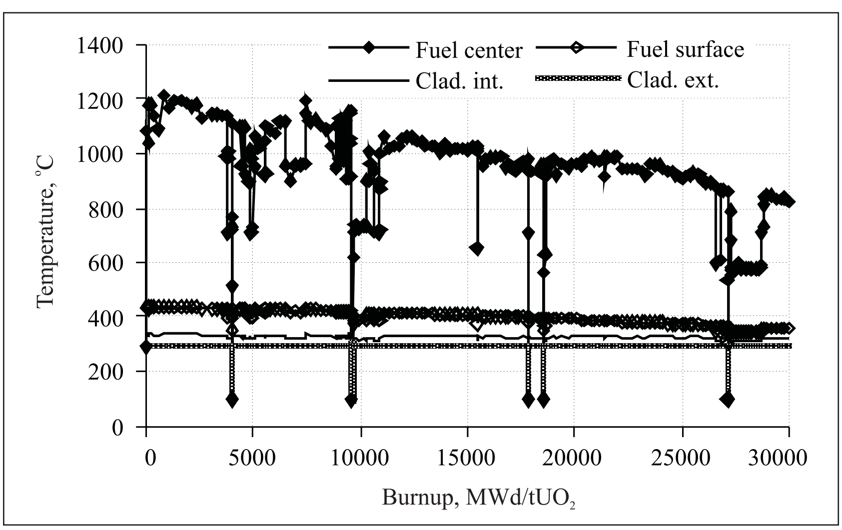

Fig. 13. Peak temperatures in fuel rod

During reactor operation, the fuel temperature decreased, but the pressure of gasses in the gap between fuel and cladding remained approximately constant (Fig. 14) because of the decreased gap between pellets and cladding (Fig. 15); the total volume of gases decreased also. The other mechanism that increased gas pressure was fission gas release. According to FEMAXI-6 analysis, the release of fission gases was constant during the whole fuel assembly operation. This means that the amount of gases was increasing. The fraction of gas mixture in the fuel rod was slowly changing (Fig. 16). The fraction of He decreased from 100% to 94%, while the fraction of Xe and Kr increased from 0 up to 6%.

Fig. 17 shows that the radius of fuel pellets was slightly increasing, while the radius of cladding was decreasing. Thus, due to radial deformations of pellets and cladding, the gap between pellet and cladding was decreasing (Fig. 15). At the moments when the reactor was shut down, the radial gap increased due to reactor cooldown. However, during reactor operation, the process of gap decrease was present. This means that fuel pellet expansion and cladding shrinking are irreversible processes. However, it should be noted that the gap between fuel pellet and cladding (segment No. 7) remained open during the whole period of normal operation (Fig. 15). The radial and axial displacements of fuel pellets were caused by increasing gas bubbles in the pellet (Fig. 18).

The elastic deformation of the fuel cladding was very small (Fig. 19). Due to temperature increase during operation up to ~300 °C at the pressure ~7 MPa from the coolant side and the pressure of gas inside the fuel rod ~1.7 MPa, the fuel cladding was compressed. Thus, the elastic deformations (Fig. 19) were caused by cladding compression (i. e. the deformations were negative). After reactor shutdown, the radial dimensions of cladding returned close to the initial conditions. The stresses of fuel rod cladding and fuel pellets, are presented in Figs. 20 and 21. As is shown in Fig. 20, the highest circumferential compression stresses were in the range of 50 MPa. These stresses were decreasing in accordance with the decreasing linear power. According to the literature [10], the yield stress for Zr + 1% Nb alloy was 180–220 MPa for 300 °C and 320–380 MPa for 20 °C.

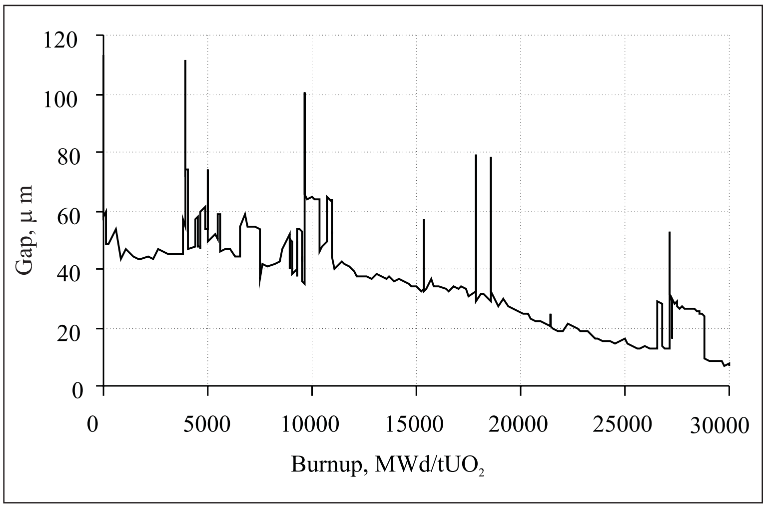

Fig. 15. Gap between pellet and cladding

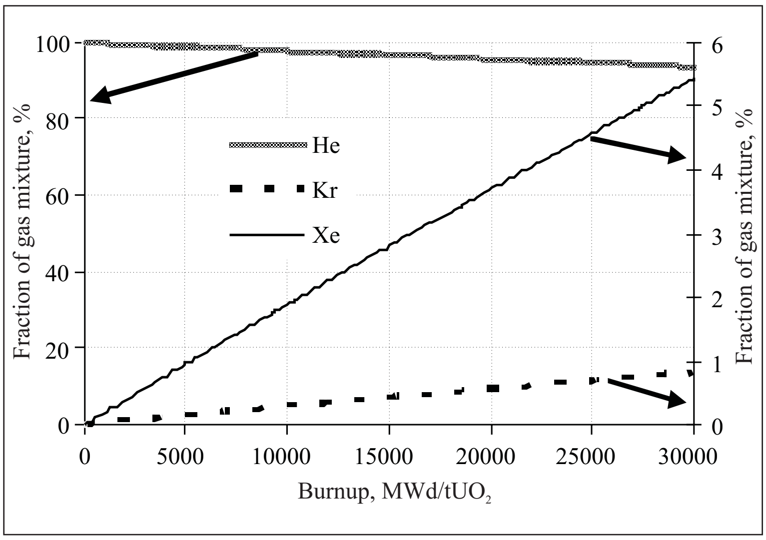

Fig. 16. Gas mixture composition in fuel rod

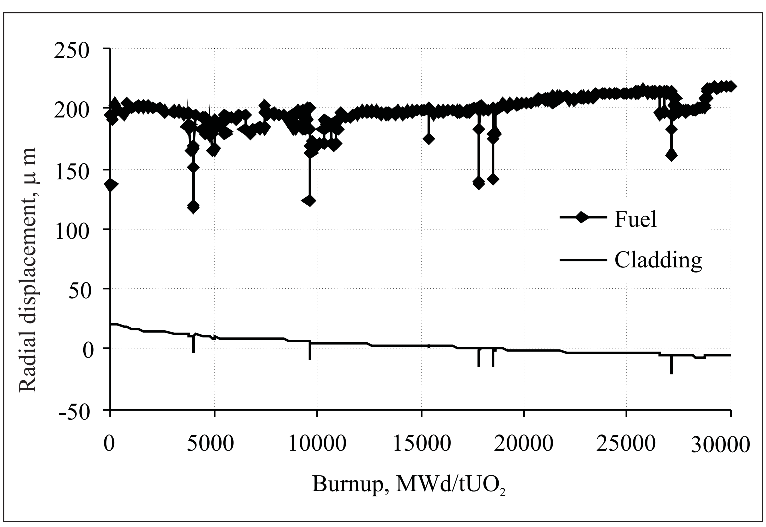

Fig. 17. Fuel and cladding radial displacement

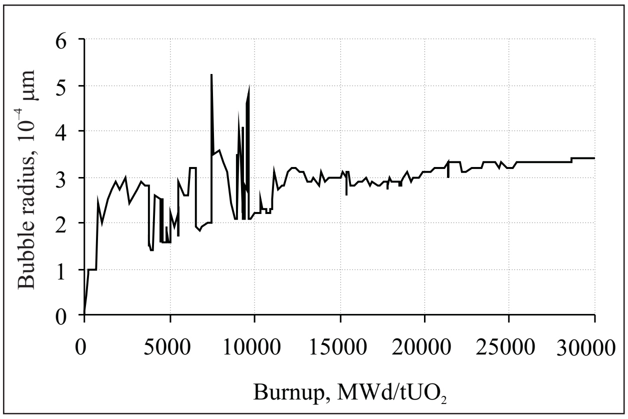

Fig. 18. Bubble radius in fuel pellet

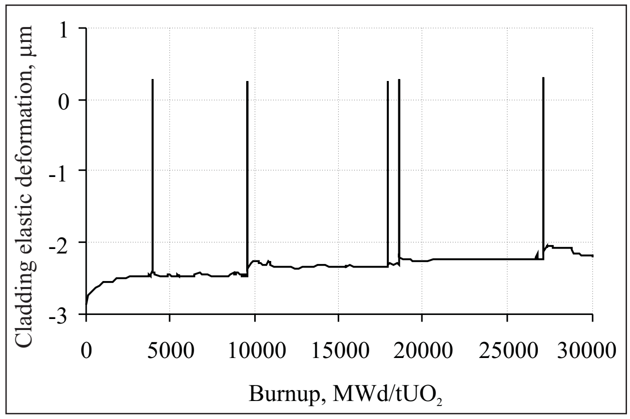

Fig. 19. Cladding elastic deformation

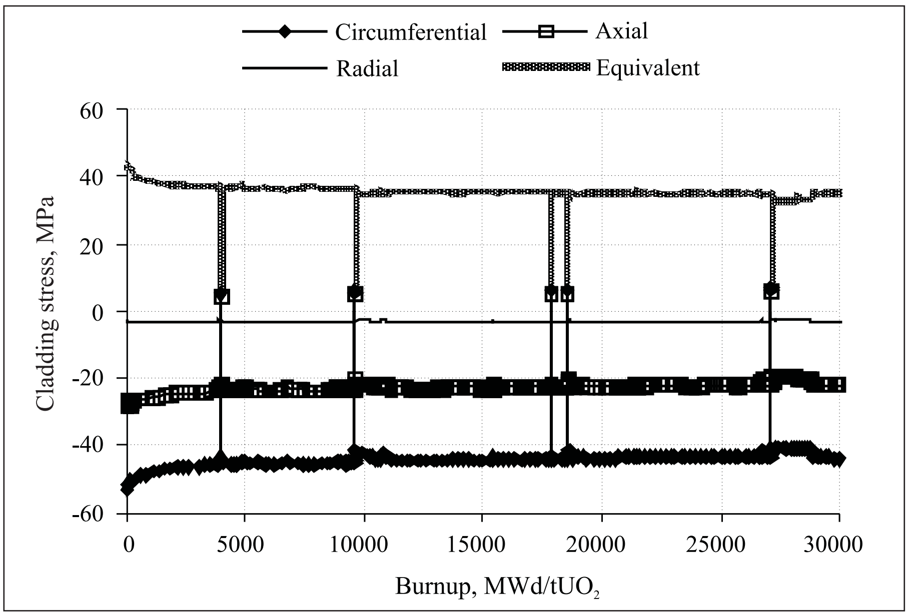

Fig. 20. Stress in fuel rod cladding

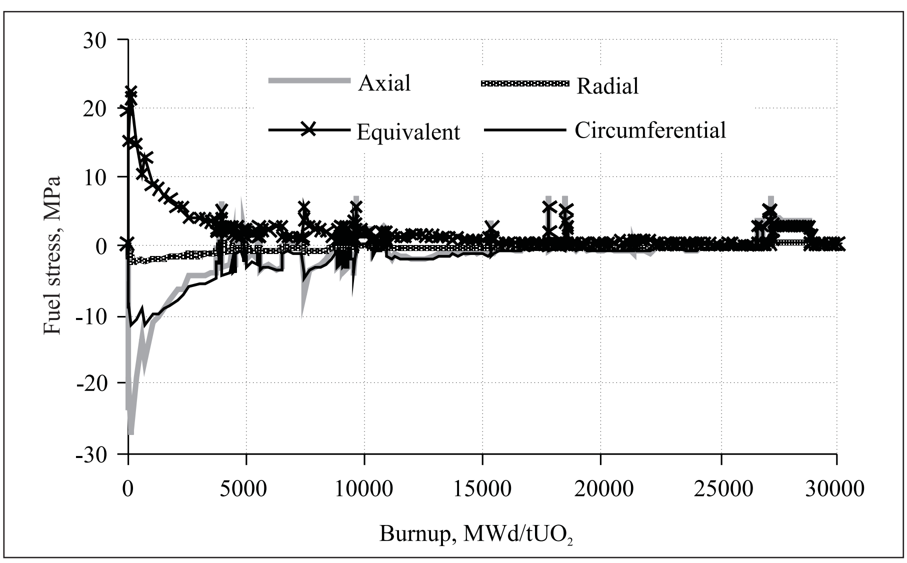

Fig. 21. Stress in the fuel pellet

After exceeding this limit of yield stresses, the fuel cladding would be affected by plastic deformation which leads to cladding failure. In our case, the calculated maximal value of circumferential compression stress in the cladding was much lower than the yield stress. Thus, the fuel cladding remained intact during the whole life of a fuel assembly.

6. SUMMARY AND CONCLUSIONS

The FEMAXI-6 code was adapted by including the thermal properties of the RBMK, and a model of RBMK-1500 fuel rod was developed. The results obtained using the developed RBMK-1500 fuel rod model and the adapted FEMAXI-6 code were compared with the results obtained at the Kurchatov Institute. A comparison of the results demonstrated a reasonable agreement.

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

1. Analysis of the influence of fuel rod nodalisation on the results showed that a fuel rod should be divided into 10 or more axial segments (11% active length of a fuel rod) to minimise the influence of nodalisation.

2. The calculation of a fuel rod during the whole life of normal operation showed that the gap between fuel pellet and cladding remained open during all the normal operation period; the elastic deformation of cladding is very small, and the maximal stresses of fuel rod cladding and fuel pellets are several times lower than the yield stresses, implying that the safety barrier was sustained, i. e. the fuel rod cladding remained intact.

3. The calculation results on the normal operation of a fuel rod could be used in future as initial conditions for simulating the processes in fuel rods stored in spent fuel pools.

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PROCESŲ, VYKSTANČIŲ RBMK-1500 REAKTORIAUS ŠILUMĄ IŠSKIRIANČIUOSE ELEMENTUOSE NORMALAUS DARBO CIKLO METU, ANALIZĖ

Santrauka

Šilumą išskiriantys elementai (ŠIEL) po eksploatacijos iš reaktoriaus aktyviosios zonos perkeliami į panaudoto kuro baseinus, tačiau prieš tai patikrinama jų būklė. ŠIEL'ų būklė gali būti vertinama ir atliekant deterministinį tyrimą. Šiame darbe, atlikus skaitinį tyrimą, buvo įvertinta RBMK-1500 reaktoriuje eksploatuoto ŠIEL'o būklė.

Iki šiol RBMK-1500 reaktoriaus ŠIEL'uose vykstančius procesus tyrinėjo RBMK reaktoriaus projektuotojai. Pastaraisiais metais Lietuvos energetikos instituto specialistai, naudodami FEMAXI-6 programų paketą, taip pat pradėjo modeliuoti Ignalinos atominės elektrinės reaktoriaus ŠIEL'uose vykstančius procesus. FEMAXI-6 programų paketas buvo pritaikytas, atsižvelgus į RBMK-1500 ŠIEL'o specifiką. Pritaikytu FEMAXI-6 programų paketu gauti skaičiavimo rezultatai palyginti su I. V. Kurčiatovo instituto specialistų gautais rezultatais. Rezultatų palyginimas parodė, kad pritaikytas FEMAXI-6 programų paketas yra tinkamas RBMK-1500 ŠIEL'uose vykstantiems procesams tirti.

Pritaikytu FEMAXI-6 programų paketu ištirti visos eksploatacijos metu ŠIEL'e vykstantys procesai. Šiam tyrimui pasirinktas ŠIEL'as iš kuro kanalo, kurio vidutinė galia 2,5 MW. Atlikus skaitinį tyrimą įvertinta ŠIEL'o būklė po eksploatacijos. Gautus rezultatus bus galima naudoti tiriant procesus ŠIEL'uose, saugomuose panaudoto kuro baseinuose.

Raktažodžiai: RBMK-1500, šilumą išskiriantis elementas, FEMAXI-6

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АНАЛИЗ ПРОЦЕССОВ В ТВЭЛАХ РЕАКТОРА РБМК-1500 ВО ВРЕМЯ НОРМАЛЬНОГО ЦИКЛА РАБОТЫ

Резюме

В конце эксплуатации должно быть определено состояние тепловыделяющих элементов (твэлов) и только тогда твэлы из активной зоны реактора могут быть помещены в бассейны выдержки топлива. Состояние твэлов может быть определено с использованием детерминистического анализа. В настоящей статье состояние твэлов РБМК-1500 после эксплуатации оценено с помощью численного метода.

До сих пор исследования процессов в твэлах РБМК-1500 проводились только проектировщиками реактора РБМК. В последние годы специалисты Литовского энергетического института начали моделировать процессы в твэлах Игналинской атомной электростанции. Код FEMAXI-6 адаптирован с учетом специфики твэлов реактора РБМК-1500. Результаты, полученные используя адаптированный код FEMAXI-6, были сравнены с результатами расчетов, выполненных специалистами института И. В. Курчатова. Совпадение результатов показывает, что адаптированный код FEMAXI-6 является подходящим для анализа процессов в твэлах РБМК-1500.

Используя адаптированный код FEMAXI-6 смоделированы процессы, происходящие в твэлах во время эксплуатации. Для этого анализа был выбран твэл из топливного канала средней мощностью 2,5 МВт. В результате численного анализа определено состояние твэла после эксплуатации. Полученные результаты могут быть использованы для анализа процессов в твэлах в бассейнах выдержки топлива.

Ключевые слова: РБМК-1500, тепловыделяющий элемент, FEMAXI-6