Advanced Fuel Cycles and Burnup Increase of WWER-440 Fuel

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1. Advanced Fuel Cycles of WWER-440

1.1. Analysis of Operational Experience of 4.4% Enriched Fuel in the Five-Year Fuel Cycle at Kola NPP, Unit 3

The year 1986 can be considered as the beginning of transition for new fuel cycles in WWER-440 reactors. That year they started using the fuel of 4.4% enrichment at Kola NPP, Unit 3. Starting with 1991 standard design fuel assemblies of 4.4% enrichment were left for the fifth year of operation at Kola NPP, Unit 3. By the year 2003, 276 fuel assemblies of 4.4% enrichment have passed through the 5-year fuel cycle at Kola 3. By now there exists extensive experience of operating fuel assemblies for four and five years up to the burnup (average in a fuel assembly) of ≈50 MWd/kgU.

Based on this, as well as on the data presented in p.2 of the report 12 fuel assemblies of 4.4% enrichment were left for the sixth year of operation (17-th fuel loading) in 2001. By the time of their discharge (the year 2002) they were operated in the reactor for 1871.4 effective days, the average burnup in the fuel assembly was ≈57 MWd/kgU.

72 more fuel assemblies were left for the fifth year of operation in 2002 as part of the 18-th fuel loading. Distribution of burnup in fuel assemblies discharged after 5 and 6 years of operation is presented in Figure 1.

Currently 36 assemblies of 4.4% enrichment are installed into the core of Kola 3 for the 6-th year of operation with the simultaneous installation of the second-generation assemblies.

1.2. Analysis of Operational Experience of Fuel Assemblies with Uranium-Gadolinium Fuel at Kola NPP, Unit 4

In 1998 at Kola 4 they started implementation of the 5-year fuel cycle with the use of uranium-gadolinium fuel (UGF) in the working fuel assemblies (WA) with the profiled average enrichment of 4.4% containing 6 gadolinium fuel rods (fuel rods

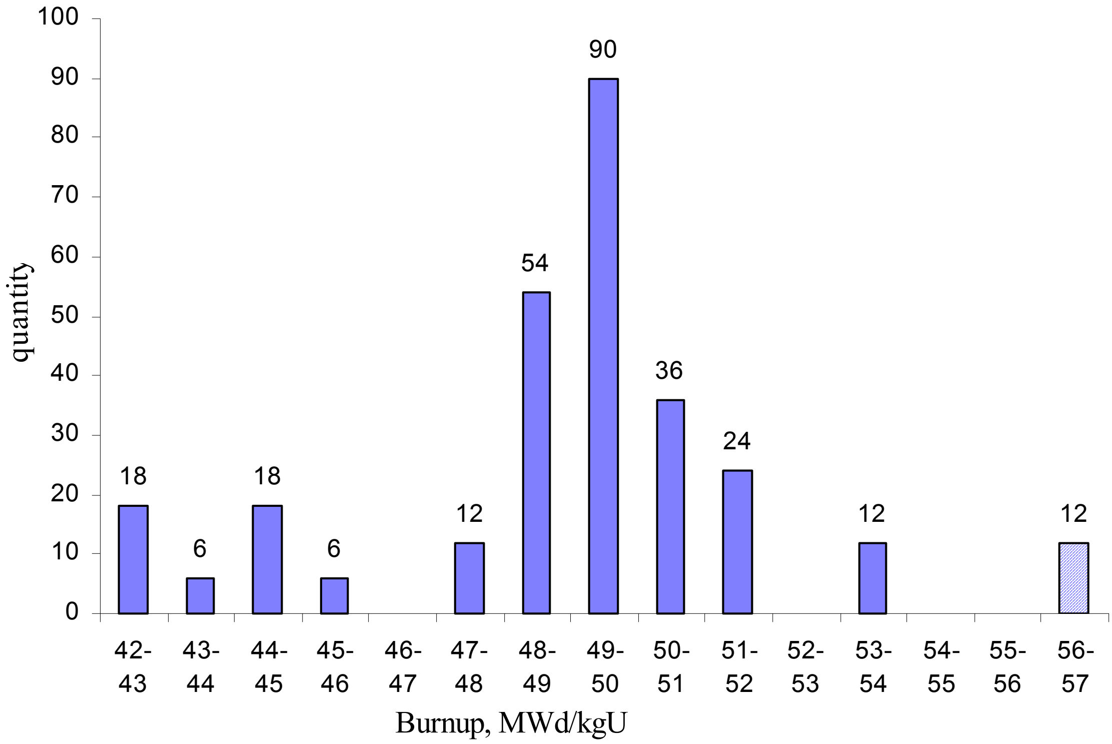

Figure 1. Dependence of number of assemblies with 4.4% enrichment discharged out of the WWER-440 reactors after 5-6 years of operation versus burnup (shaded indicates assemblies that operated for 6 years)

in which Gd_2O_3 burnable absorber is integrated into fuel). Currently (the 16-th fuel loading) 168 WA with UGF are operated in the core of Kola 4. 12 of these WA are being operated for the fourth year. After 3 years of operation (2002) maximum burnup in these assemblies was 38.66 MWd/kgU. All of the assemblies are still leak tight.

In the core of Kola 4 during the previous 15-th fuel loading there were 102 WA with UGF, of which: 12 WA were in operation for the 3-rd year, 24 WA were in operation for the 2-nd year, 66 WA were in operation for the 1-st year.

No criteria for the pre-term termination of operation of fuel assemblies in the core of the reactor regarding the coolant activity were reached. Analysis results of experimental data of WA with UGF, and computer prognosis allow performing further operation of Unit 4 with UGF at the nominal power level.

1.3. Implementation of Advanced 5-Year Fuel Cycle with the Assemblies of Second Generation at Kola NPP, Unit 3

Since 2002 at Kola 3 they are implementing advanced 5-year fuel cycle with fuel assemblies of the second generation. They differ from the assemblies of the first generation in:

- A larger step in positioning fuel rods in a fuel bundle (from 12.2 to 12.3 mm);
- Longer fuel columns in fuel rods (for control rod and fuel assemblies {CA}OW type – from 2320 to 2360 mm, for WA VW type – from 2420 to 2480 mm);
- Less diameter of the central hole in fuel pellets (from 1.6 to 1.35 mm);
- Less outer diameter of fuel rods (from 9.1 to 9.07 mm);
- UGF;
- Hafnium plates in the docking unit of the CA;
- Size of shrouds for fuel assemblies of VW and OW type is 145 mm.

Table 1 lists main neutronic characteristics of fuel loadings for the steady-state modes of fuel reloadings in the reactor of Kola 3 for the case of implementation of the 5-year fuel cycles with the use of 4.4% enriched WA and fuel assemblies of the second generation with UGF. In 2002 (18-th fuel loading) a new set of the second generation assemblies was loaded at Kola 3:

- 54 WA with the average enrichment of 4.25%;
- 6 CA with the average enrichment of 3.82%.

1.4. Experimental-Industrial Operation of Control Assemblies with Upgraded Docking Unit at NV NPP, Unit 4

Due to the implementation at NPPs with WWER reactors of the fuel cycles with the increased burnup depth, and therefore, due to the additional design restrictions related to the permissible values of the linear heat rate jumps per fuel rod, as well as to the perspective use of manoeuvre modes of the reactor operation the docking unit of the CA of WWER-440 was upgraded. Hafnium plate was introduced into the design of the docking unit of these assemblies in order to decrease the peaks of energy deposition in the fuel rods of the peripheral rows of the neighbouring WA. The first set of such CA (12 assemblies) with the average enrichment of 3.82% was loaded into the core of NV NPP 4 in the year 2000 (26-th fuel loading). Currently such CAs are being operated in the reactor cores at NV NPP 4 and Kola 2 and 3.

2. Operability of WWER-440 Fuel under High Burnup

Reaching the burnup level as high as 60-66 MWd/kgU in a fuel rod is expected for WWER-440 fuel due to the transition to five- and six-year fuel cycles. Therefore, forecasting of characteristics of fuel rod behavior at this burnup level is important. Let us present some of the data used for justification of possibility of trial operation of assemblies described above in p.1.1. Justification was performed on the basis of:

- Generalization of the data of operation and of post-reactor investigations of WWER-440 fuel rods and fuel assemblies [1-8] having operated within five years (including comparison of these data with the data for fuel rods of lower burnup) and forecasting of operability of WWER-440 fuel rods intended for operation for six years.
- Calculation justification of operability of these fuel rods.

By the present time more than 7 million fuel rods have been operated in WWER-440 reactors. The reached burnup levels were higher than 50 MWd/kgU on the average per fuel assembly. It shall be noted that the average fraction of the damaged fuel rods was $(1\text{-}1.3)\cdot10^{-5}$ per one fuel cycle during the time period of 1993-1998, but in some years it was as high as $5\cdot10^{-6}$ [2]. This can be explained by the design and technical approaches used for designing of fuel rods, and firstly by presence of the central hole in the fuel pellets and facets at the pellet ends, as well as by the properties of zirconium alloy used for fuel cladding fabrication. This alloy has demonstrated good corrosion resistance and mechanical strength. Fuel rods of the WWER type were and are tested in MP, MIR, and HBWR reactors in the framework of a number of experimental programs (SOFIT-1, RAMP, FGR-1&2, IFA-503.1&2, etc.).

For WWER fuel rods there exists a large set of data on post-reactor investigations, and on testing of spent fuel in research reactors and high-temperature testing facilities. Up to the year 1987

Table 1. Neutronic characteristics of fuel loadings for the steady-state modes of fuel reloadings in the reactor of Kola 3 for the case of implementation of the 5-year fuel cycles with the use of 4.4% enriched WA and fuel assemblies of the second generation with UGF

№	Parameter	Fuel cycle 1 steady-state	Fuel cycle 2, steady-state	
			odd	even
1.	Reactor thermal power, [MWt]	1375		
2.	Number of fuel assemblies in the core	349		
3.	Geometry of fuel assemblies FA)	1.5 mm 2.1 mm	shroud wall thickness of WA – 1.5 mm, shroud wall thickness in CA – 1.5 mm	
4.	Structural material of spacer grids	Zirconium		
5.	Profiling of fuel enrichment in the cross-section of a fuel assembly	VW and OW type of fuel assemblies are used		
6.	Type of the used burnable absorber	-	Gd_2O_3	Gd_2O_3
7.	Schematics of fuel reloading	With reduction of neutron leak		
8.	Number of fresh FA loaded during reloadings in respect to the types	78 66(K), 12(H)	66 12(OW), 54(VW)	66 6(OW), 60(VW)
9.	Average enrichment of fresh fuel, [%]	4.28	4.17	4.21
10.	Operation time in the reactor of discharged FA, [years] Average Maximum	 4.47 5	 5.38 6	 5.20 6
11.	Average burnup of discharged fuel, [MWd/kgU] - for all FA Maximum fuel burnup, [MWd/kgU] - in a FA - in a fuel rod - in a fuel pellet	 45.76 (50.4) 50.5 (57.7) 58.2 (65.1) 66.6 (74.1)	 49.89 52.37 59.50 66.70	 49.02 52.07 60.10 67.70
12.	Duration of the reactor operation between fuel reloadings, [eff. days]	319	295	300
13.	Maximum irregularity in the power of FA, Kq^{max}	1.34	1.39	1.37
14.	Average relative power of peripheral FA, $\overline{K}q$ BOC EOC**	 0.47 0.51	 0.39 0.46	 0.40 0.47
15.	Maximum value of the linear heat rate of fuel rods ql^{max}, [W/cm]*	295.5	272.5	276.4
16.	Irregularity in power of fuel rods Kr	1.55	1.55	1.55
17.	Effective specific consumption of natural uranium, [kg/MWd]	0.211	0.193	
18.	Effective specific scope of separation work, [swu/MWd]	0.126	0.115	

* With consideration of the margin coefficient $K_{marg.} = K_N \times K^{table}_{engin} = 1.04 \times 1.12 = 1.165$,

** End of operation period by exhaust of the reactivity margin at power of 100%.

Note: In p.11 of Table 1 as applied to the five-year fuel cycle 1, values of burnup presented in the parenthesis correspond to the burnup level reached when 12 assemblies were left for the sixth year of operation.

fragments of fuel rods and elements of fuel assemblies were studied in the protection chambers. Since 1987 after putting into operation of a specialized facility for the post-reactor investigations of fuel assemblies 8 fuel assemblies of WWER-440 and 20 of WWER-1000 were [1] investigated (altogether about 7000 fuel rods) with the maximum burnup in a fuel rod up to 56.8 MWd/kgU. Refabricated fuel rods were also tested in MIR reactor till sufficiently high levels of burnup. This level of tests representativeness allows to have a clear understanding of the state of fuel rods and fuel assemblies at the average per fuel assembly burnup up to 50 MWd/kgU and higher after operation in WWER reactors.

Operational experience, data of post-reactor investigations and calculations allow to forecast good operability of WWER fuel rods in case of the increase of the average burnup in a fuel assembly up to 55-60 MWd/kgU with good confidence. The data of post-reactor investigations indicate that along with the increase of the burnup:

- Mechanical properties of the claddings do not practically change and remain at a sufficiently high level;
- Cladding deformation (both in radial and axial directions) including due to the fuel-cladding interaction should not result in exhaust of operability of fuel assemblies;
- Crack development is not observed in the claddings, accumulation of damages in the claddings should not result in exhaust of operability of fuel assemblies;
- Corrosion and hydrogenation of claddings should not result in significant deterioration of operability of fuel assemblies;
- Gas release from the fuel and reduction of the value of free space under the fuel cladding should not lead to the situation when the pressure under the fuel cladding exceeds the coolant pressure.

2.1. Brief Generalization of the Data of Post-Reactor Investigation of WWER-440 Fuel Assemblies and Rods [1-8]

Post-reactor investigations demonstrated that WWER-440 fuel assemblies do not practically change their sizes and shapes up to the average burnup of 50 MWd/kgU: the fuel bundle remains integral, positioning of fuel rods does not change. Let us present generalized results of the post-reactor investigations of WWER-440 fuel rods (see for example [1]).

Elongation of fuel rods (under normal conditions after being discharged from the reactor). Fuel rods elongate proportionally to the burnup with the rate of ≈0.1% per 10 MWd/kgU. No deviations from this dependence are observed to the burnup of ≈50 MWd/kgU. It is possible to assume that for the average burnup in a fuel assembly up to 55-60 MWd/kgU elongation of fuel rods of this design will not exceed the rated value (25 mm).

Change of the fuel rod diameter (under normal conditions). Up to the burnup of 35-40 MWd/kgU:

- Fuel rod diameters decrease due to radiation creep of claddings induced by the coolant pressure. At high burnup levels the rate of diameter decrease slows down. This can be explained by the beginning of the fuel-cladding interaction (PCI). At the burnup of ≈43 (38-45) MWd/kgU diameter starts to increase. This can be explained by intensive fuel-cladding interaction following the pattern of the “rigid” contact, and by the fact that fuel starts interacting with the cladding practically along the total height.

At the burnup higher than ≈43 MWd/kgU:

- Under the influence of swelling fuel the cladding starts to increase its diameter due to radiation creep. In the most burnt sections corrugation (“bamboo effect”) of the claddings with the pitch aliquot to the length of fuel pellets and the amplitude up to 50 μm is observed. Metallographic studies demonstrate that coordinates of the cladding deformation peaks coincide with the coordinates of the pellet ends.

Maximum decrease of the cladding diameter is 60-90 μm at the burnup of 35-50 MWd/kgU. At a higher burnup cladding diameter starts to increase. Therefore, decrease of the diameter does not reach the values of 120-150 μm, at which worsening of mounting of fuel rods in spacer grids (SG) is possible. SGs are located as designed with no displacement and distortion against the fuel bundle. No violation in the parallel positioning of fuel rods was observed. Fuel rods are in the contact with the SG: fuel rod stripping and pulling forces are larger than the zero, and as a rule are equal to higher than 20 and 10 kilogram-force, respectively.

Fuel-cladding gap. The fuel-cladding gap decreases with the increase of burnup, and at the burnup of ≈40 MWd/kgU is fully reduced under operating conditions. At the burnup of ≈50 MWd/kgU the gap is equal to 0-10 μm under normal conditions.

Fuel cladding corrosion. As a rule external surface of all the studied fuel rods is covered with the uniform oxide film less than 8 μm thick. The film is tightly adhered to the main metal. Thickness of the oxide film reaches 10-12 μm in the zone of welds.

At the burnup level of up to ≈35-40 MWd/kgU oxide film can be seen at the internal surfaces of fuel claddings. Thickness of this film can change from 0 to 10 μm even in one cross-section of the cladding (this can be accounted for by fuel-cladding axial asymmetry due to the fuel cracking). At the burnup higher than ≈45 MWd/kgU under conditions of total fuel-cladding contact oxide film is more uniform along the perimeter and can be as thick as 15 μm. Sometimes 15 μm thick interaction layer is observed on the cladding internal surfaces. This layer contains mixtures of U-(Pu)-Cs-O or Zr-Cs-O type.

Weak dependence of thickness of oxide films on the WWER fuel claddings versus burnup allows to assume that no change of the film thickness is to be expected at higher burnup levels.

Hydrogenation of fuel claddings. Small amount of lamellar zirconium hydrides not exceeding 100 μm in size is contained in the claddings of some fuel rods. Hydrogen content in the irradiated claddings of all leak-tight fuel rods is ≈ (3-8)·10^{-3}% mass.

Mechanical properties of WWER-440 and WWER-1000 fuel rods are practically the same for all the tested assemblies and do not depend of either burnup or the place along the fuel rod height, where the sample was cut out from. In the analyzed range of average burnup (from 13.1 to 50.5 MWd/kgU per fuel assembly) mechanical properties are characterized with high strength and sufficiently high plasticity. Such stability of mechanical characteristics of Zr1%Nb alloy is explained by the ordered arrangement of dislocation loops, formation of which is completed at relatively low neutron fluences with the energy higher than 0.4 MeV - ≈10^{19} n/cm^2.

This makes it possible to state that the fuel cladding of Zr1%Nb alloy will maintain its high mechanical properties at the burnup higher than 55 MWd/kgU per fuel assembly.

Macro- and microstructure of fuel after operation of up to five years can be described as follows:

- Pellets are fragmented into four and more pieces (mainly by radial cracks), but maintain initial configuration. Diameter of the central hole is practically unchanged;
- Average grain size in the center of the pellet is unchanged;
- At low burnup (up to ≈3 MWd/kgU) fuel density increases due to radiation densification. Fuel starts to swell at high burnup. When the burnup in a fuel pellet is 63 MWd/kgU swelling reaches the value of ≈4% vol.

When the burnup is higher than 45 MWd/kgU along the cross-section:

- Closing of cracks (or sections of radial cracks) is observed at the periphery of fuel pellets. This can be accounted for by the increased swelling in this area due to higher burnup at the periphery of the pellet compared with the average burnup along the cross-section (by ≈70%);
- A clearly distinguished area with the changed microstructure starts to develop at the fuel pellet boundary. The so-called rim-layer is of a higher burnup, higher plutonium content, and higher porosity. This layer is characterized by the presence of a lot of small gas bubbles, disappearance of the initial grain structure, and formation of the new sub-grains with much less dimensions (less than 1 μm). At the burnup of ≈63 MWd/kgU width of this layer can reach 150 μm.

Fission gas release (FGR) from the fuel increases with the increase of burnup. At low burnup levels FGR is low. For fuel rods with high fuel temperature (with high linear heat generation rate) FGR is also dependant on the value of this temperature. Sufficiently sharp growth of FGR starts at the burnup (in the fuel rod cross-sections) higher than 45-50 MWd/kgU, which corresponds to the burnup per fuel rod higher than 40 MWd/kgU. Growth of release is explained by supersaturation of the matrix with fission products and surface rim-effect. This growth is of an athermanous character and takes place even at relatively low fuel temperature.

For the tested WWER-440 fuel rods FGR from the fuel is approximately 0.4-1% off the generated fission products up to the burnup of 40-45 MWd/kgU, and it grows furthermore at higher burnup. The maximum FGR was 6% from FA-222 fuel rod.

Free space in the fuel rod (under normal conditions) is reduced in the process of burnup. This can be accounted for by the decrease of the cladding diameter due to radiation creep (up to the burnup of ≈45 MWd/kgU), and by the fuel swelling. The minimum value of free space for the tested WWER-440 fuel rods was 8.1 cm^3 for the fuel rod from FA-222.

Gas pressure inside the cladding (under normal conditions) for WWER-440 fuel rods at low burnup correspond to the filling pressure – 0.5-0.7 MPa. With the increase of burnup value of this pressure increases, which can be accounted for by the reduction of free space, and by the increase of the amount of gas inside the cladding due to FGR to gas medium. At the burnup of 35-50 MWd/kgU pressure is 0.8-1.4 MPa. The maximum measured pressure of gas medium inside the fuel cladding was 1.84 MPa for FA-222 fuel rod that had abnormally high value of FGR – 5.2%. Pressure inside claddings of WWER-440 fuel rods under normal operating modes is to be no more than ≈3-3.5 times higher than the pressure after reloading (under normal conditions). Therefore, it is possible to forecast that pressure of the gas medium inside the cladding will not exceed coolant pressure up to the burnup in the fuel rod of ≈65 MWd/kgU even in case of growth of FGR.

Analysis of the state of welds, of SGs, central tube, and spring unit of the tested fuel rods and assemblies also allows to predict operability with the increase of burnup.

[8] presents results of the post-reactor investigations of two WWER-440 fuel assemblies after being operated for a five-year fuel cycle: FA-222 [4-7] and CA that was successfully operated in the core of NV NPP 4. After being discharged this CA had the calculated burnup of 50.48 MWd/kgU;

maximum calculated value of the average burnup per fuel rod was 60.15 MWd/kgU. Besides we used results of the post-reactor investigations of the FA#РП-3, 3-77A that was operated in the reactor of NV NPP 3 for five years. The burnup of this assembly after discharge was 48.95 MWd/kgU. The results of inspection demonstrated that all of the fuel rods were leak-tight.

Conclusion made after generalization of the data of the post-reactor investigations of WWER-440 fuel rods: Data of post-reactor investigations of fuel rods of the assemblies that have been operated for five years and have reached the average burnup of ≈50 MWd/kgU, and comparative analysis of these data with the data of post-reactor investigations of the assemblies of less burnup demonstrate absence of drastic degradation of operability of WWER-440 fuel rods, when the burnup per fuel assembly gets as high as ≈50 MWd/kgU. Along with the growth of burnup no exponential changes of parameters determining operability of fuel rods was observed. This allows to conclude that it is possible to increase burnup of WWER-440 fuel rods up to the average burnup level per assembly of ≈55-60 MWd/kgU.

2.2. Data of Calculation Justification of Operability of WA Fuel Rods Intended for the Operation for Six Years

It was planned to leave 12 WA for the 6-th year of operation. 17 most loaded fuel rods were selected of these WA for the further calculation analysis. Sampling was performed on the basis of BIPR-7A and PERMAK computer analyses.

TOPRA-1 [7,9] computer code for thermal-physical analysis was used in the computer analysis. TOPRA-1 code is intended for modeling of the behavior of fuel or fuel-gadolinium rods of the WWER type in the quasi-steady-state mode. Calculations were performed for four options in respect to the initial parameters of fuel, cladding, and fuel rod as a whole and covering all of the potential states of real fuel rods.

Preliminary study of the analyses results has demonstrated that of the thermal-physical criteria it is necessary to meet only the criterion of not exceeding coolant pressure by the maximum pressure inside the cladding. Calculations with the margin coefficients of 1.04 in respect to the linear hear generation rate and burnup were performed for the fuel rods of the most dangerous regarding violation of this criterion option (set of geometrical and structural parameters). The most conservative options regarding the models of FGR and the model accounting for PCI were used in the calculations. That means that the presented calculation results are of a conservative nature in respect to FGR, pressure inside the fuel cladding, and increase of the fuel rod diameter after formation of the fuel-cladding contact.

Let us present some of the calculated data for the maximum burnt fuel rod (65.3 MWd/kgU per fuel rod) for the option of the “average input parameters”. To demonstrate this for the above fuel rod Figures 2-4 present:

- Maximum and average linear heat generation rates (without margin coefficient) – Figure 2;
- Maximum and average fuel temperatures in a fuel rod – Figure 3;
- Pressure of the gas medium inside the cladding and FGR – Figure 4,

as the function versus the time of operation.

Calculated were the maximum: fuel tempera-

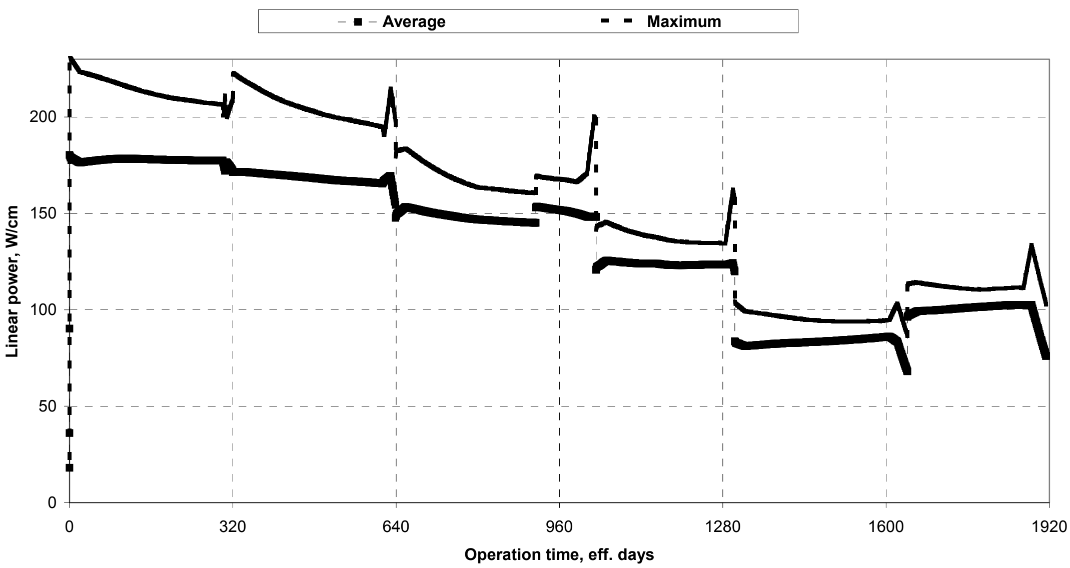

Figure 2. Dependence of the maximum and average heat generation rates of the fuel rod versus the time of operation

ture – 958˚C, FGR from the fuel – 9.15%, pressure inside the cladding – 7.37 MPa. Pressure inside the cladding after the discharge under normal conditions was 2.71 MPa. For this fuel rod of the maximum-unfavorable option regarding the initial effective fuel-cladding gap with consideration of the 1.04 margin coefficient in respect to the power and burnup the calculated maximum: fuel temperature was 1053˚C, FGR from the fuel – 13%, pressure inside the cladding – 11 MPa.

3. Analysis of the Obtained Results

Average change of the radius of the fuel cladding outer surface at the core section (under normal conditions) became positive for three fuel rods. Here, in some axial zones positive increase of this radius was up to 18 μm. In the central (along the height) most burnt sections radius of the fuel cladding became larger than the initial one under operating conditions by 30 μm (in average along the height – by 16 μm).

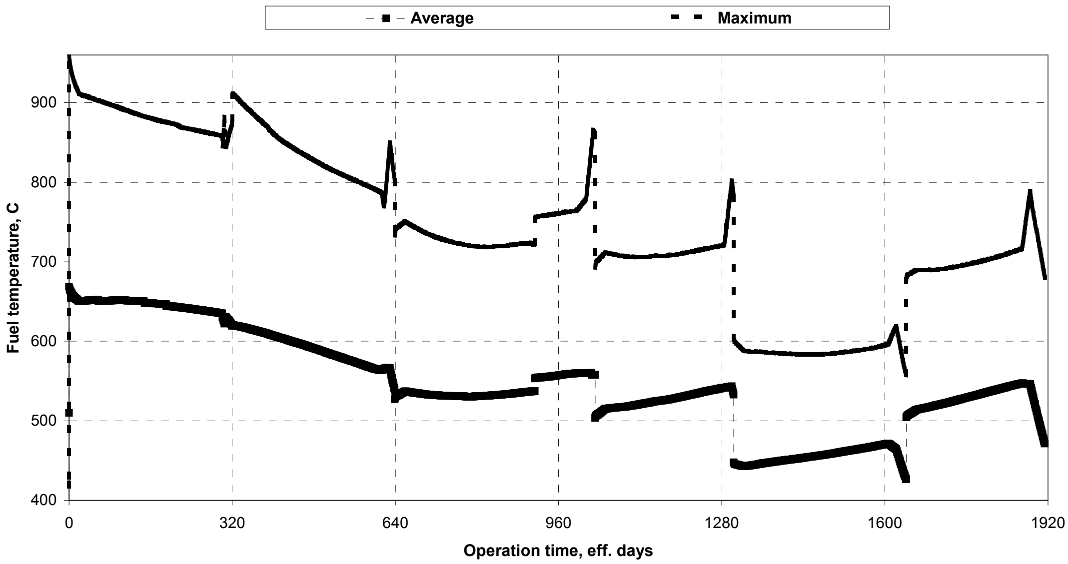

Figure 3. Calculated dependencies of the maximum and average fuel rod temperatures versus the time of operation

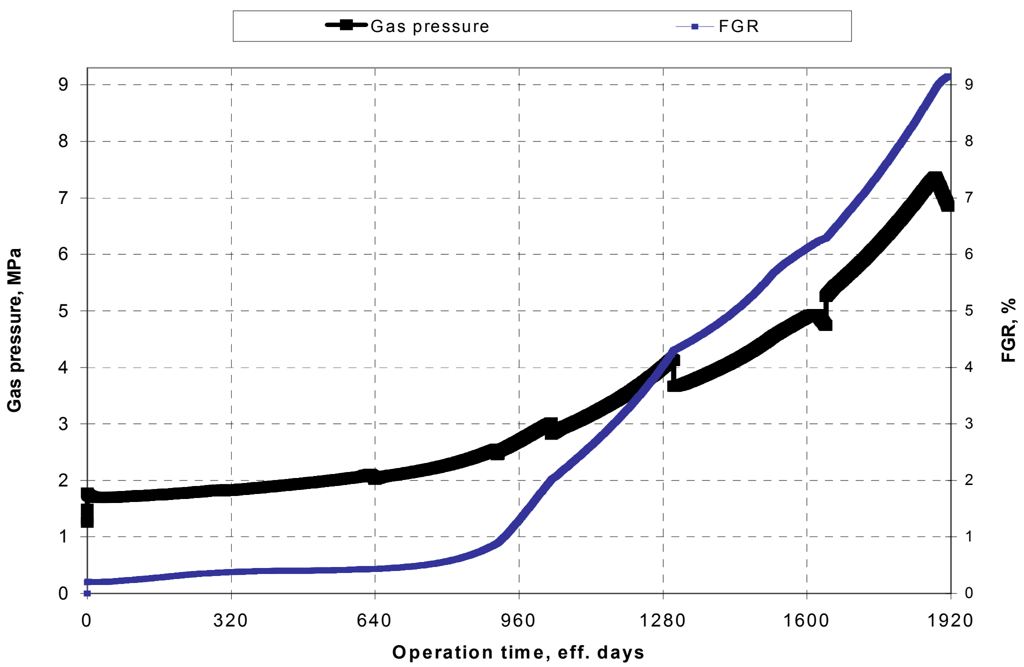

Figure 4. Calculated dependence of the gas medium pressure inside the fuel cladding and FGR versus the time of operation

Elongation of fuel claddings after discharge from the reactor grows proportionally to the fuel burnup. Maximum value of elongation – 14.5 mm is with a significant margin lower than the maximum allowable value. Calculations of the fuel strength parameters were performed by TOPRA-2 code [9]. Calculated results have demonstrated that after the fuel-cladding contact is reached cladding elongation rate slows down and gets less than the value presented in p. 2.1, i.e. 01% per 10 MWd/kgU. That means that after beginning of PCI the rate of fuel cladding elongation does not grow (due to expansion by the swelling fuel), it goes down. Decrease of the rate is caused by the decrease of the creep rate in the axial direction due to less neutron flux and cladding temperature (at low linear heat generation rates characteristic of highly burnt fuel rods). Besides, decrease of the rate is also caused by the negative contribution to the cladding axial deformation, due to the positive plastic hoop strain. It should be noted that the models of the cladding creep and PCI of the used option of TOPRA-2 have not been completely verified yet.

In connection with the obtained fact indicating decrease of the elongation rate it should be noted that the average value of the experimentally measured elongation for all fuel rods of WA-222 (burnup of 49.33 MWd/kgU) is by ≈ 4.7% less than the same value for all fuel rods of WA-198 (45.88 MWd/kgU). Results calculated [10] by TRANS-URANUS code for the fuel rods of WA-222 also demonstrate some slow down of the cladding elongation rate at the burnup higher than ≈46 MWd/kgU. Figure 5 presents elongation of fuel rods after discharge (under normal conditions) calculated by TRANS-URANUS [10], as well as the results of post-reactor investigations of the fuel rods of WA-198 and WA-222.

On the whole maximum fuel temperature decreases in the process of operation. This is caused by both reduction of the linear heat generation rate, and by the processes in the fuel rod. Maximum fuel temperature for the reviewed fuel rods did not exceed 1100˚C.

For the reviewed fuel rods characteristic by the relatively low temperature but high burnup, FGR is mainly of the athermanous nature and does not exceed 14%. Growth of the gas pressure inside the cladding is due to the reduction of free space in the fuel rod and FGR. It occurs even in case of the decrease of the fuel rod average linear heat generation rate. For the reviewed fuel rods pressure of the gas medium inside the cladding does not exceed 11 MPa, i.e. even with:

- Consideration of overlapping of the margin coefficients in respect to the power and burnup;
- The use of conservative values of the input parameters, and
- The use of comparatively conservative model for FGR,

it stays less than the coolant pressure.

Of the mechanical criteria of the fuel rod operability it should be noted that PCI according to the scheme of the “rigid” contact (fuel-cladding contact with the selected cracks in the fuel) for the fuel rod of the minimum gap option occurs at the burnup of ≈30 MWd/kgU. With such burnup levels under the conditions of the relatively low linear heat generation rates and of non-exceeding the limiting values

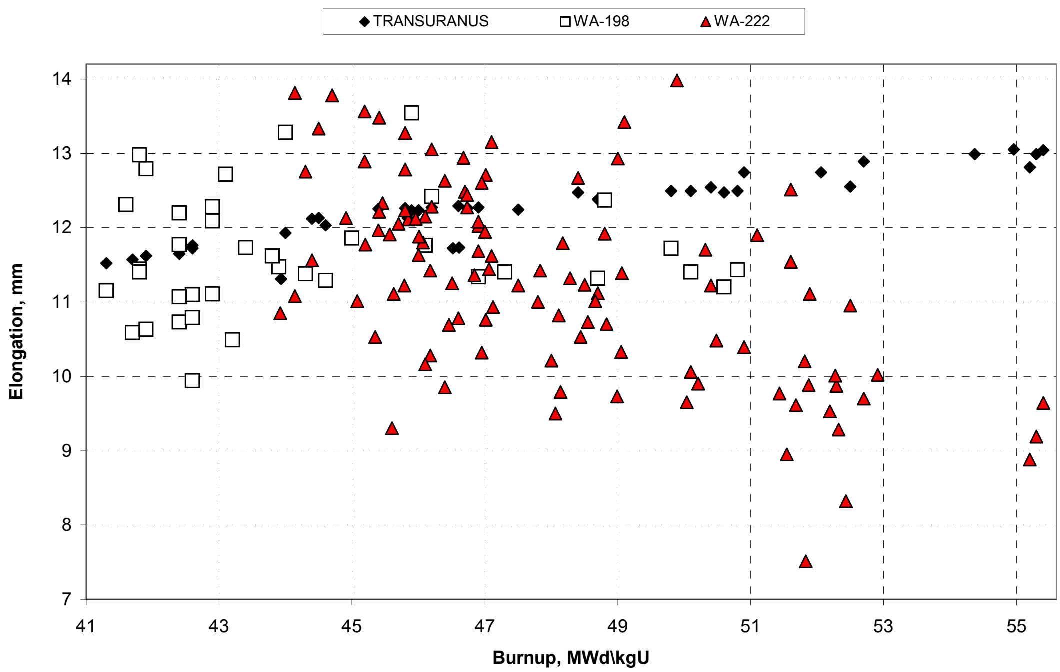

Figure 5. Data on elongation of WWER-440 fuel claddings

of the power drops (as the function of the local burnup) this interaction does not lead to exhaust of operability of fuel rods.

The obtained results indicate that fuel rods of WWER-440 assemblies intended for operation within six years of the reviewed fuel cycle totally preserve their operability. Performed analyses have demonstrated possibility of the fuel rod operability during the fuel cycle.

12 assemblies were loaded into the reactor unit of Kola 3 in 2001 (p. 1.1). The predicted burnup in six assemblies was 59.2 MWd/kgU. Calculated values of the burnup after operation for WA were ≈57 MWd/kgU, for fuel rods – up to ≈61 MWd/kgU. Data on the coolant activity, specific activity of the benchmark iodine radionuclides of the reactor primary circuit, control of the integrity of fuel rods of the assemblies that were operated for six years indicate that not a single assembly has reached the criterion for the early discharge.

4. Acknowledgements

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