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Reactor physics and safety aspects of various design options of a Russian light water reactor with rock-like fuels

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Abstract

This paper presents results of analytical studies on weapons grade plutonium incineration in VVER (640) medium size light water reactors using a special composition of rock-like fuel (ROX-fuel) to assure spent fuel long-term storage without its reprocessing. The main goal is to achieve high degree of plutonium incineration in once-through cycle. In this paper we considered two fuel compositions. In both compositions weapons grade plutonium is used as fissile material. Spinel ($MgAl_2O_4$) is used as the 'preserving' material assuring safe storage of the spent fuel. Besides an inert matrix, the option of rock-like fuel with thorium dioxide was studied. One of principal problems in the realization of the proposed approach is the substantial change of properties of the light water reactor core when passing to the use of the ROX-fuel, in particular: (i) due to the absence of ^{238}U the Doppler effect playing a crucial role in reactor's self-regulation and limiting the consequences of reactivity accidents, decreases significantly, (ii) no fuel breeding on one hand, and the quest to attain the maximum plutonium burnup on the other hand, would result in a drastical change of the fuel assembly power during the lifetime and, as a consequence, the rise in irregularity of the power density of fuel assemblies, (iii) both the control rods worth and dissolved boron worth decrease in view of neutron spectrum hardening brought on by the larger absorption cross-section of plutonium as compared to uranium, (iv) β_{eff} is markedly reduced. All these distinctive features are potentially detrimental to the reactor nuclear safety. The principal objective of this work is that to identify a variant of the fuel composition and the reactor layout, which would permit neutralize the negative effect of the above-mentioned distinctive features.

1. Input data and analysis technique

The analysis has been made for a VVER-640 reactor, taken as an example [1]. The main characteristics of the reactor are presented in Table 1. This reactor design has been developed by the OKB 'Gidropress' using VVER-1000 reactor as a basis. Reactor vessel and fuel assembly (FA) designs are taken similar to that of VVER-1000 reactor but power density in the core is rather low. VVER-640 is referring to the new generation reactors that assure high safety level. For instance probability per reactor of the core melting has been evaluated as $3.2 \times 10^{-7}\ a^{-1}$.

Two uranium-free rock-like fuel (ROX) compositions were studied: (a) weapons grade plutonium oxide in an inert matrix, and (b) weapons grade plutonium oxide in a matrix of thorium dioxide. The characteristics of fuel compositions are presented in Table 2. Results of analysis of reactor neutronics for the following three options of the FA loading are presented:

Option 1: 1100% loading of ROX with inert matrix;
Option 2: 100% loading of ROX with a ThO_2;
Option 3: heterogeneous loading (44% FAs with inert matrix ROX-fuel and 56% FAs with 4 wt% ^{235}U UO_2 fuel).

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Table 1
Input data on reactor and fuel assembly (FA)

Characteristics	Values
Reactor thermal power, N_{th} (MW)	1800
Core height, H_{core} (cm)	353
Core equivalent radius, R_{core} (cm)	158.19
Coolant pressure in the core (MPa)	15.7
Coolant flow rate (kg s^{-1})	10 235
Cladding dimensions, $d \times \delta$ (mm)	9.1×0.69
Cladding material	99%Zr + 1%Nb
Fuel	ROX/UO_2
Number of cells in FA	331
Number of absorber rod cells in FA	18
Number of FA in the core	163
FA lattice spacing in the core (cm)	23.6
Guide tube dimensions, $d \times \delta$ (mm)	13×1
Material	99%Zr + 1%Nb
Cladding dimensions, $d \times \delta$ (mm)	8.2×0.6
Material	Steel
Absorber	B_4C
Density, ρ_{B_4C} (g cm^{-3})	1.8

Table 2
Fuel compositions

Fuel	Components (fraction/wt%)
(a) ROX-fuel, $\bar{\rho} = 4.8$ g cm^{-3}	PuO_2 (10.1); ZrO_2 (48.5); Y_2O_3 (13.5); $MgAl_2O_4$ (27.4); Er_2O_3 (0.5)
(b) ROX-fuelz + ThO_2, $\bar{\rho} = 6.3$ g cm^{-3}	PuO_2 (6.5); ThO_2 (61.5); $MgAl_2O_4$ (32.0)

Excess reactivity compensation in the option of pure ROX-fuel is performed by erbium added as a burnable poison.

Reactor characteristics were evaluated using ACADEM a three-dimensional diffusion code [2]. Preparation of neutronics data for the ACADEM code was made using the WIMSD5 code [3] and the Nuclear Data Library developed at the SSC RF – IPPE.

Table 3
The main characteristics of fuel loading options

Characteristics	Options		
	1	2	3
Number of annually loaded ROX per UO_2 FA	33	26	12/24
Heavy atoms loading in FA, G_{FA} (kg)	21.3	233.7	20.8/ 400.6
Pu loading in ROX-fuel FA, G^{Pu}_{TBC}	21.3	22.1	20.8
Refueling interval, Δt_{eff} (d)	271.3	279.0	300.0
FA life time, t_{eff} (d)	1546	1949	1800/ 1200
Energy production of fuel			
$\overline{E}$ for heavy atoms (MW d kg^{-1})	748	88.6	738/ 36.5
$\overline{E}$ per assembly (MW d)	15 711	20 688	15 350/ 14 819
Max value of FA power non-uniformity coefficient $\mathscr{K}_q$	1.34	1.34	1.35
Max value of non-uniformity of power density, $\mathscr{K}_V$	1.50	1.53	1.60
Initial boric acid concentration (ppm)	1320	1290	1080
Worth of working group of control rods at BOC (%)	0.62	0.57	0.72
Total worth of control rods (%)	6.3	6.4	7.15
Annual reactor loading (kg a^{-1})			
Plutonium	719	574	263
Thorium	0	5500	0
Uranium (4.0% ^{235}U)	0	0	9700
Annual reactor unloading (kg a^{-1})			
Plutonium	87	74.6	15.2
Thorium	0	5220	0
Uranium-238	0	0	9315
Uranium-233	–	95	–
Fraction of burnt plutonium (%)			
In ROX-fuel	88	87	94
In the whole reactor	88	87	57

2. Results of neutronic analyses

The main results of neutronic analyses performed for the three options of fuel loading are presented in Table 3. Analysis was made in a four-group approximation for the following neutron energy values in the lower group: 0.183 MeV; 367.26 eV; 1.02 eV and 0.0. As follows from the table approximately equal amount of weapons grade plutonium (about 20 kg) was loaded into the ROX-fuel FAs in all options. Core full power lifetime is within the range of 270–300 d. In all loading options coefficient of FA power rating non-uniformity (K_q) is lower than 1.35, which is acceptable from the standpoint of Russian regulatory requirements.

It follows from Table 3 in the first option maximum amount of weapons grade plutonium is irradiated annually (719 kg). About 88% of loaded plutonium is incinerated. The highest plutonium burnup in ROX-fuel assemblies is achieved in option 3 (hybrid loading). However production of new plutonium in uranium fuel assemblies results in the decrease of total plutonium amount in the third option only by 57%. Option 2 also provides high effectiveness of weapon grade plutonium incineration, namely: 87% of loaded amount. However in this option annual yield of ^{233}U is 95 kg. In our opinion this option can be considered not as the option of eternal disposal of spent fuel but as that of long-term fuel storage with postponed reprocessing.

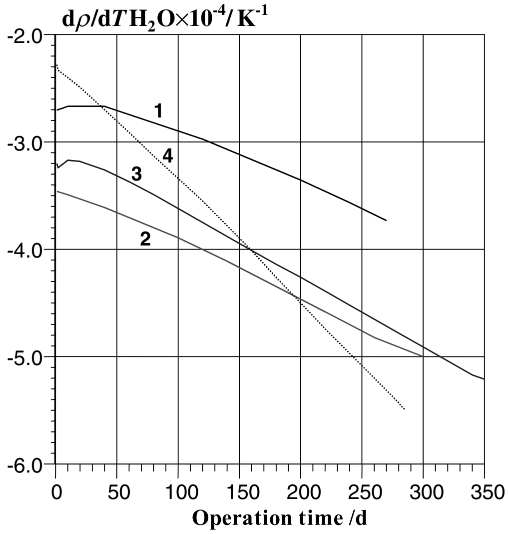

Fig. 1. Moderator temperature reactivity coefficients versus effective power operation time (curves 1 to 3 – respectively, 1 to 3 loading options, 4 – standard UO_2 loading).

Change of reactivity coefficients (ρ) caused by water and fuel temperature and boric acid concentration, and change of the β_{eff} value in the course of fuel operation are shown in Figs. 1–4. Curve numbers 1, 2 and 3 relate to the first, second and third loading options. Curve 4 corresponding to the full core loading with standard uranium fuel is given for comparison.

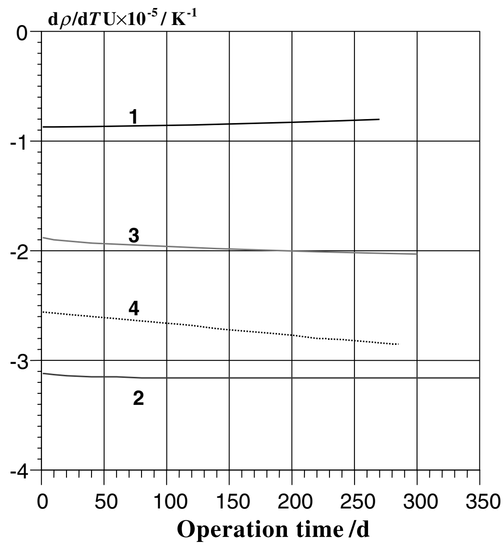

Fig. 2. Fuel temperature reactivity coefficients versus effective power operation time (curves 1 to 3 – respectively, 1 to 3 loading options, 4 – standard UO_2 loading).

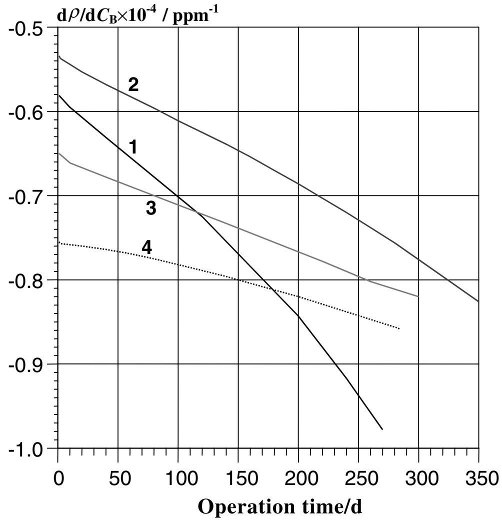

Fig. 3. Boron concentration reactivity coefficients versus effective power operation time (curves 1 to 3 – respectively, 1 to 3 loading options, 4 – standard UO_2 loading).

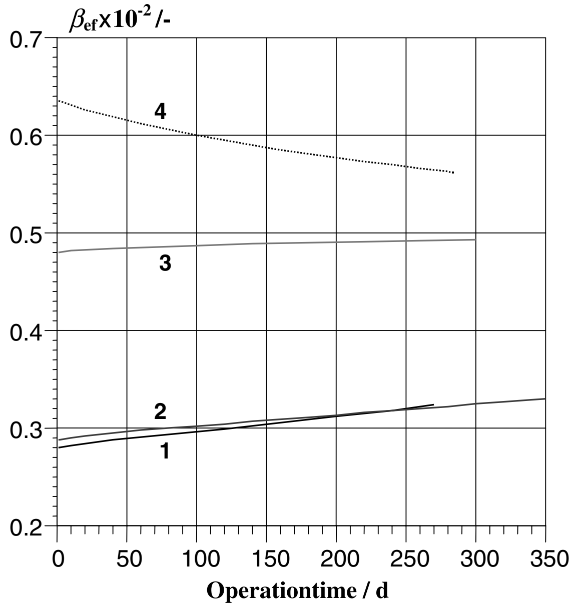

Fig. 4. Delayed neutron fractions versus effective power operation time (curves 1 to 3 – respectively, 1 to 3 loading options, 4 – standard UO_2 loading).

As follows from Fig. 2 fuel temperature reactivity coefficient for pure ROX-fuel option 1 is about three times lower than that for standard uranium loading. It can be noted that option 2 (full loading with ROX-fuel and thorium addition) is preferable in this case. It could be forecasted that in uranium-free options (1 and 2) effective delay neutron fraction is much less than that for standard loading.

3. Reactivity-initiated accidents

The analysis of two reactivity related accidents were made for the above fuel loading options. The following initial events were considered: (a) uniformly accelerated ejection of the control rod group during 0.1 s from its position at 284 cm above the core bottom without scram, and (b) unauthorized withdrawal of the control rod group at 2 $cm\,s^{-1}$ velocity from the same position. Accident analysis was made for the beginning and the end of the fuel lifetime.

ROX-fuel element design is similar to that of a UO_2 fuel element, with the fuel-cladding gap width being equal to 0.1 mm. In contrast to traditional fuel, it is assumed that there is no central hole in the fuel pellet. Data on thermal conductivity and heat capacity of ROX-fuel provided by the designer are presented in Table 4. It follows from the table that ROX-fuel has rather high thermal conductivity as compared to that of UO_2 fuel. For a conservative analysis of the reactivity-initiated accidents lower values of the ROX-fuel thermal conductivity were used (see Table 5). This analysis was carried out for the most complicated case with full loading of the ROX-fuel in the reactor core (see option 1*, Table 6)

The thermal expansion coefficient of both samples of ROX-fuel was assumed to be equal to 14.5×10^{-6} K^{-1} within the 873–1073 K temperature range. The initial gas pressure in the fuel-cladding gap was 2 MPa. The elasticity modulus and Poisson coefficients for ROX-fuel were assumed to be equal to those for traditional fuel.

The accident process modeling was made using the three-dimensional dynamic code DYN3D/H1.1 [4]. The main results of the analysis are presented in Table 6. As follows from these data, the maximum achievable temperature of the ROX-fuel in the above accidents is well below its melting point ($\sim$2200 K), and the minimum margin to dry-out (departure from nucleate boiling ratio – DNBR) is significantly over 1. As it was expected maximum reactor power peak is observed in the first option with pure ROX-fuel. However this does not cause overheating of fuel and cladding. In spite of low value of Doppler effect, which is much less than that in options 2 and 3, this accident is not critical for option 1. High thermal conductivity of fuel composition assures low initial temperature value and intensive heat removal to the coolant. The latter makes it possible to assure negative reactivity feedback on the coolant temperature on the early accident stage, thus preventing fuel overheating.

It should be stressed that the calculations are based on the preliminary results of measurements of thermophysical properties of non-irradiated ROX-fuel, specifically, on high value of fuel thermal conductivity (see Table 4). If the realistic value of thermal conductivity is essentially lower or decreases as a result of irradiation in neutron flux (see the conservative date in Table 5), the maximum fuel temperature will be 200–300 K higher. But even in this case thanks to low specific power rate in the VVER-640 reactor core (64.9 $kW\,l^{-1}$) the maximum fuel temperature will not reach the melting point. Other characteristics are also in the acceptable range.

In option 3 the highest fuel and cladding temperatures and lowest DNBR value are observed for uranium FAs. However these values are kept within permissible limits.

4. Accident caused by the rupture of the main circuit pipeline

An accident caused by the rupture of the main circuit pipeline (MCP) of the reactor plant was studied for the third fuel loading option. Consequences of this event are rapid loss of a major amount of coolant in the circuit, and deterioration of heat removal from the core that

Table 4
Thermophysical properties of ROX-fuel (with unpublished IPPE κ data)

T (K)	573	673	773	873	973	1073	1173	1273
κ ($W\,m^{-1}\,K^{-1}$)	14.4	14.8	15.1	15.2	15.4	15.4	15.5	15.5
C_p ($J\,g^{-1}\,K^{-1}$)	0.86	0.97	1.07	1.17	1.27	1.38	1.48	1.58

Table 5
Thermophysical properties of ROX-fuel with conservative κ data e.g. Spinel data from Ref. [6]

T (K)	473	673	873	1073	1273	1473	1673	1773
κ ($W\,m^{-1}\,K^{-1}$)	4.39	3.94	3.77	3.53	3.38	3.15	3.09	3.06
C_p ($J\,g^{-1}\,K^{-1}$)	0.86	0.97	1.07	1.17	1.27	1.38	1.48	1.58

Table 6
Extrema values of parameters under conditions of reactivity related accidents

Parameters	Time point of the core lifetime	Option 1	Option 1*	Option 2	Option 3	
					ROX	UO_2
Ejection of control rods group						
Peak power-rated power ratio	Beginning	2.32	2.39	1.49	1.44	
	End	2.05	2.10	1.61	1.58	
Max fuel temperature (K)	Beginning	1158	1430	1022	1124	1456
	End	1059	1221	953	995	1307
Maximum cladding temperature (K)	Beginning	623.1	623.0	622.7	622.8	322.7
	End	622.6	622.6	622.4	614.9	617.0
DNBR[a]	Beginning	1.31	1.32	1.32	1.48	1.49
	End	1.54	1.55	1.53	1.96	1.84
Unauthorized withdrawal of control rods group						
Peak power-rated power ratio	Beginning	1.24	1.23	1.10	1.14	
	End	1.23	1.22	1.10	1.14	
Max fuel temperature (K)	Beginning	1147	1428	1017	1121	1456
	End	1059	1218	953	995	1307
Maximum cladding temperature (K)	Beginning	623.1	623.0	622.7	622.8	622.7
	End	622.6	622.6	622.4	614.9	617
DNBR[a]	Beginning	1.31	1.32	1.31	1.48	1.49
	End	1.54	1.55	1.53	1.96	1.84

* Conservative analysis.
[a] Departure from nucleate boiling ratio.

might cause a significant increase of the fuel element cladding temperature. For this reason, according to regulatory documents on safety, it should be justified that the fuel element cladding temperature and the degree of local oxidation in this accident do not exceed maximum permissible values, which are equal to 1473 K and 18% respectively.

4.1. Accident scenario

The conditions and conservative assumptions of the accident scenario include:

(1) Initial rated parameters of the reactor plant takes into account possible deviations of
  - 4% in the reactor thermal power,
  - 5% in the mass flow rate of the primary coolant through the core,
  - 2 K in the core inlet temperature.
(2) Instantaneous double-ended break of the MCP cold leg is assumed (MCP inner diameter is 620 mm).
(3) The process of reactor shut down is initiated after the primary coolant pressure drops under 14.7 MPa. It continues for 4 s until all of the scram control rods are completely inserted into the core.

Table 7
Accident chronology

Time (s)	Events
0.0	Guillotine-type rupture of the cold leg of reactor circuit (initial event)
0.02	Generation of signal for reactor shutdown because of pressure decrease in the primary circuit
0.32	Start of insertion of safety rods
4.1	Maximum temperature of the fuel element cladding is achieved
4.32	Completion of safety rods insertion
8.3	Switching on DHRS[a] high pressure hydraulic accumulators
127.6	Switching off DHRS[a] high pressure hydraulic accumulators
214.0	Start of coolant supply to the reactor from atmospheric pressure tanks of DHRS
231.5	Coolant outflow from the reactor is completely compensated by DHRS
300.0	End of calculation

[a] Decay heat removal system.

(4) Postulated failures of safety systems are
   – failure of one of the emergency core cooling system (ECCS) pressurized accumulators,
   – the failure of all ECCS pumps (active components).

4.2. Analytical tools and reactor plant calculation model

The accident model was made using the DYN3D/RELAP5 code package [4,5] capable of modeling an accident process taking into account a change of the three-dimensional distribution of the core power density, being the sum of prompt neutron power and decay heat. Each FA has its own hydrodynamic channel. In order to determine the maximum temperature of the fuel elements, the 'hot channel' model was used. This model is capable of determining temperature conditions of the fuel element taking into account possible deviations of its parameters and calculation errors. It was assumed for the analysis that the 'hot channel' power is 25% higher than the fuel element power averaged over the FA radius. The sequence of events of this accident is shown in Table 7, and the main results of analysis are presented in Figs. 5–7.

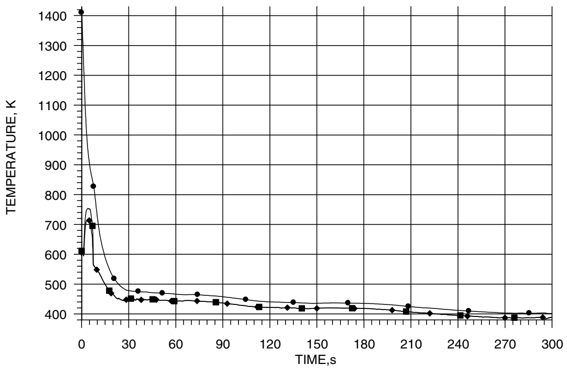

Fig. 6. Fuel rod maximum temperature (● fuel centerline, ■ cladding).

It follows from the results that the maximum rate of coolant outflow from the primary circuit can be as high as $\sim$27 000 kg s^{-1} shortly after the pipeline rupture. As

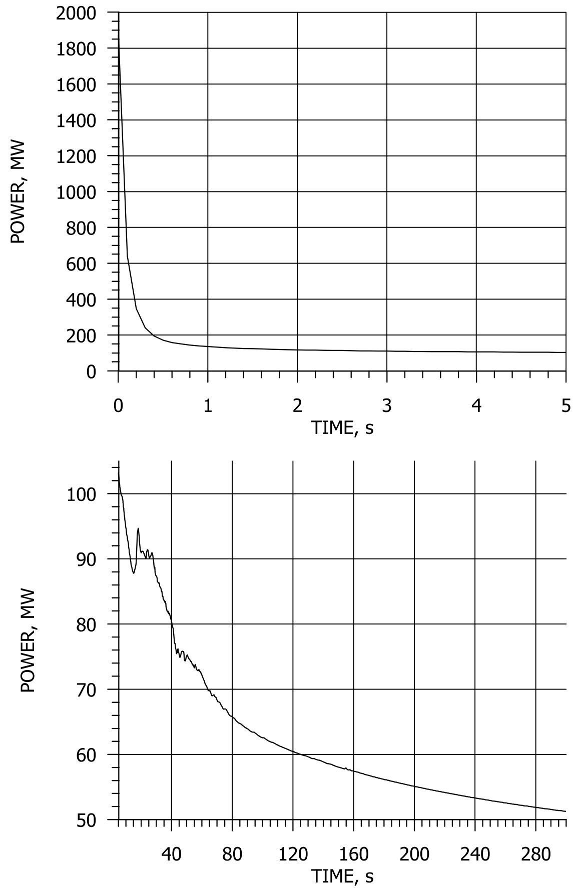

Fig. 5. Reactor total power as a function of time.

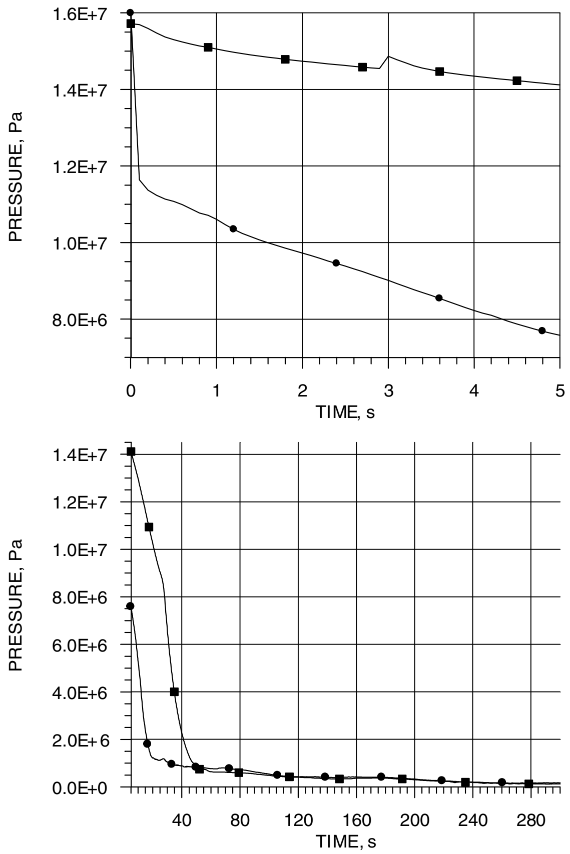

Fig. 7. Reactor pressure (● upper plenum, ■ pressurizer).

a result of this, the reactor pressure decreases rapidly (Fig. 7), and about 0.02 s after the initial event, a signal for a reactor scram is generated. However, the reactor power starts to decrease before the scram, i.e. within 0.32 s after the accident start (Fig. 5). This power change is caused by the negative reactivity feedbacks as a result of decrease of coolant density in the core.

The maximum cladding temperature of a 'hot' fuel element can be as high as ~750 K (Fig. 6). This fuel element belongs to a UO_2 fuel assembly. ROX-fuel element cladding temperatures do not exceed this value. No significant oxidation of cladding material occurs in the course of the accident.

Continuous filling of the core with borated water from the decay heat removal system (DHRS) starts at ~235 s.

Results of the accident analysis show that all requirements of regulatory documents, as far as fuel element cladding parameters are concerned, are reliably met (with a considerable margin).

5. Conclusion

Analysis has shown that ROX-fuel can be used in principle for plutonium utilization in the VVER-640 reactor. Various fuel loading options with inert matrix fuel were considered. These options are meeting all requirements of nuclear safety and design limits for both normal operation and accident conditions.

The highest rate of plutonium utilization (719 $kg\,a^{-1}$) is reached in the first option with 100% ROX-fuel with an inert matrix. In this case, ~88% of loaded plutonium is destroyed. Low Doppler reactivity coefficient is not an obstacle for this option introduction in the VVER-640 even if ROX-fuel thermal conductivity appeared to be essentially lower than the value used in our main calculations.

As regards core neutronics, in our opinion, the second option with thorium addition is preferable. However, another fissile isotope ^{233}U is produced in this case, thus contradicting the original ROX-fuel concept, which implied direct eternal disposal of spent fuel. In this case, long-term reliable disposal of spent fuel is supposed, its reprocessing being postponed for several decades or even centuries.

In the third option with mixed loading, the maximum degree of weapons grade plutonium disposal is achieved (up to 94% in ROX-fuel subassemblies). At the same time, new reactor grade plutonium is produced in the uranium fuel assemblies. This option may be useful for weapons grade plutonium disposal if ROX-fuel is loaded into operating reactors.

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