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A study on the physical characteristics of a supercritical light-water reactor loaded with (^{233}U-Th-^{238}U) oxide fuel

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Abstract

The paper investigates a possibility and attractiveness of using (U-Th)-fuel in light-water reactors with super-critical coolant parameters. It is proposed to dilute ^{233}U with ^{238}U to enhance the proliferation resistance of this fissionable isotope. Various fuel compositions are analyzed and compared on fuel breeding, achievable values of fuel burn-up and cross-sections of parasitic neutron absorption. Two structural materials are considered as a fuel cladding: zirconium-based alloy and stainless steel.

1. Introduction

For some years a series of international initiatives has been put forward, where the medium-term and long-term prospects of nuclear power development were considered. One of these initiatives, the “Generation-IV International Forum”, was proposed and sponsored by the USA. In 2002 the Consultative Committee on Nuclear Power Investigations prepared a “technological plan of actions” in respect of the Generation-IV nuclear power systems. As a consequence, six innovative concepts of reactor technologies and associated nuclear fuel cycle technologies [1] were selected for further joint studies and development.

One of these six concepts is a light-water reactor with super-critical coolant parameters, i.e. high-temperature reactor cooled by high-pressure coolant under operation conditions above the thermodynamics critical point, or **S**uper-**C**ritical **L**ight-**W**ater **R**eactor (SCLWR). The SCLWR investigations have already started, and it is expected that these reactors could be commissioned over the next three decades.

Some studies [2,3] considered the use of enriched uranium and MOX-fuel in SCLWR. The present paper is devoted to studying neutron-physical parameters of SCLWR loaded with dioxide (U-Th)O_2 fuel (**homogeneous** mixture of ^{232}Th dioxide with ^{233}U and ^{238}U dioxides). The capabilities of various fuel compositions were analyzed from the standpoint of achieving high fuel burn-up. It is noteworthy that dilution of ^{233}U with ^{238}U for enhancing proliferation resistance of this fissionable isotope was put forward for the first time by the well-known American physicist and participant of the Manhattan Project Dr. T. Taylor [4].

2. Specific features of Super-Critical Light-Water Reactors

In Super-Critical Light-Water Reactors highly-pressurized (up to 25 MPa) light water plays the role of coolant, which is pumped upwards through the reactor core. Under such high pressure, light water transits into a super-critical state, which allows it to reach a significant difference of coolant temperature along the height of the reactor core and, as a consequence, to reach high thermal efficiency of the nuclear power plant (up to 45%). This possibility is one of the basic advantages offered by SCLWR-type reactors.

Presently, there are two basic concepts of SCLWR. The first concept, in a certain sense, may be regarded as a traditional one, i.e. fuel rods are assembled in a regular lattice and coolant is pumped up in the inter-rod space.

However, because of very high coolant temperature at the outlet from the reactor core and, as a consequence, low coolant density, the upper part of the reactor core contains a relatively small amount of hydrogen, the main neutron moderating material. So, a resonance neutron spectrum forms in this part of the reactor core. If such a neutron spectrum is undesirable for any reasons (for example, from the standpoint of neutron multiplying properties), then another SCLWR concept may be used. The second SCLWR concept is based on the following feature: in the upper part of the reactor core some fuel rods contain a neutron-moderating material (zirconium hydride or light water, for instance) instead of fuel. In this case, even in the upper part of the reactor core, where the density of the main moderator (light water) is low, the softened neutron spectrum is formed due to the introduction of an additional moderator. A disadvantage of the second concept in comparison with the first is a lower quantity of fuel rods in the same volume of the reactor core, i.e. a lower specific energy generation rate per volume unit. The present paper considers the first SCLWR concept.

The following shortcomings can be found in this concept of SCLWR: "… extremely large increase of coolant enthalpy and relatively low heat capacity of overheated steam results in high sensitivity to any non-uniformities of coolant heating. Large variations in density of coolant-moderator can cause considerable non-uniformity in spatial distribution of heat generation rate. High coolant temperature requires the use of alloys with high content of nickel as cladding materials of fuel rods. A substantial problem of the uniflow reactor is a problem to provide acceptable temperatures of fuel rod cladding" [2].

Some measures have been proposed in Ref. 2 for overcoming the above disadvantages. For example, micro fuel elements in direct contact with coolant flow could be used in SCLWR. A two-pass scheme of coolant motion was considered in [3]. At first, coolant moves downwards in the peripheral region of the reactor core. Coolant flows are mixed in the bottom collector and the coolant is then directed upwards in the central part of the reactor core for further heating.

As mentioned above, there is a large difference between outlet and inlet temperatures of coolant (consequently, between outlet and inlet coolant density) in the SCLWR core. Appropriate inlet and outlet values are as follows: from 280°C (0.78 g/cm^3) at inlet to 500°C (0.09 g/cm^3) at outlet of the reactor core. In addition, water density sharply decreases in the temperature interval around 380°C. Temperature dependence of water density (pressure – 25 MPa) is presented in Fig.1.

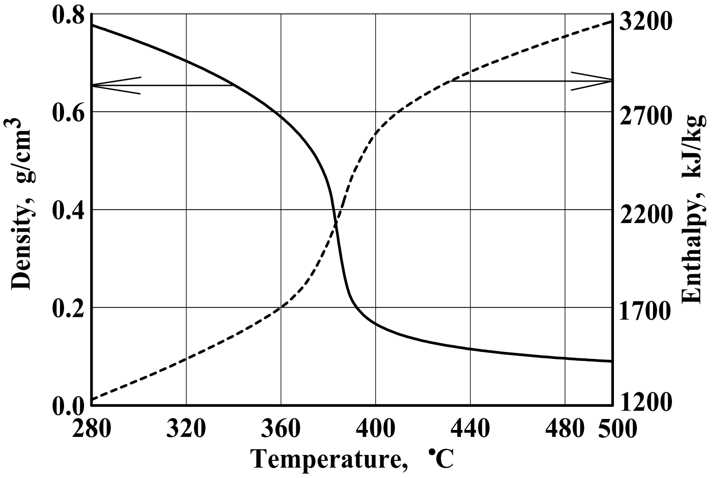


Fig.1. Properties of water under pressure of 25 MPa

The growth of enthalpy defines, in essence, the amount of heat received by the coolant. Water enthalpy also undergoes sharp non-uniform variations within the SCLWR operation range of coolant temperature. The temperature dependence of water enthalpy (pressure – 25 MPa) is also presented in Fig.1. It follows from the curve that the most intense cooling of fuel rods (the largest variation of water enthalpy under heating per centigrade) takes place in the middle part of the SCLWR core.

So large and non-uniform variations of water properties within the temperature range of SCLWR operation can result in large variations of the fuel neutron-physical properties. Therefore, it seems reasonable to divide the reactor core into several axial regions for further detailed neutron-physical studies. In the present investigation the following three axial regions of SCLWR core were selected: inlet region (average water density γ_{H2O} = 0.72g/cm^3), central region (γ_{H2O} = 0.4g/cm^3) and outlet region (γ_{H2O} = 0.1g/cm^3).

3. Neutron-physical properties of ^{233}U, ^{239}Pu, ^{232}Th and ^{238}U in SCLWR neutron spectrum

Dioxide fuel compositions (mixtures of ^{233}U, ^{232}Th and ^{238}U dioxides) were considered with fixed content of ^{233}U and various contents of ^{232}Th and ^{238}U. Neutron-physical parameters of these fuel compositions were determined using the SCALE-4.3 computer code package and ENDF/B-V evaluated nuclear data file [5]. Ratios of spectrum-averaged micro cross-section of neutron radiative capture to fission cross-section ($\alpha = \sigma_c/\sigma_f$) are presented in Table 1 for the following values of water density and for the following fuel compositions: density - 0.72g/cm^3, 0.4g/cm^3 and 0.1g/cm^3; ^{232}Th-based composition (4% ^{233}U + 96% ^{232}Th), ^{232}Th-^{238}U-based composition (4% ^{233}U + 48% ^{232}Th + 48% ^{238}U) and ^{238}U-based composition (4% ^{233}U + 96% ^{238}U).

As it follows from these data, the α values for all fissionable isotopes under analysis weakly depend on fuel composition as well as on coolant density. These values belong to the following ranges:

$$\alpha\,(^{233}\mathrm{U}) \approx 0.12 \div 0.14$$
$$\alpha\,(^{239}\mathrm{Pu}) \approx 0.55 \div 0.65$$

The α value of ^{239}Pu is larger than that of ^{233}U by a factor about 4, though the number of fission neutrons produced by ^{239}Pu is remarkably larger than that by ^{233}U:

$$\nu_f\,(^{233}\mathrm{U}) \approx 2.52$$
$$\nu_f\,(^{239}\mathrm{Pu}) \approx 2.89$$

In fact, in SCLWR neutron spectra, the neutron-multiplying properties of ^{233}U are superior to those of ^{239}Pu for all fuel compositions and water densities under consideration in terms of effective secondary neutron amounts:

$$\nu_{\mathrm{eff}} = \frac{\nu_f}{1+\sigma_c/\sigma_f} = \frac{\nu_f}{1+\alpha} \Rightarrow \begin{cases} \nu_{\mathrm{eff}}\left(^{233}\mathrm{U}\right) \approx 2.22 \\ \nu_{\mathrm{eff}}\left(^{239}\mathrm{Pu}\right) \approx 1.82 \end{cases} \tag{1}$$

Table 1

Ratio of neutron capture cross-section to fission cross-section, $\alpha = \sigma_c/\sigma_f$

Fraction in fuel (%)		$\gamma_{H2O} = 0.72\mathrm{g/cm^3}$		$\gamma_{H2O} = 0.4\mathrm{g/cm^3}$		$\gamma_{H2O} = 0.1\mathrm{g/cm^3}$	
^{232}Th	^{238}U	^{233}U	^{239}Pu	^{233}U	^{239}Pu	^{233}U	^{239}Pu
96	0	$\frac{7}{59}=0.12$	$\frac{70}{123}=0.57$	$\frac{4.6}{35}=0.13$	$\frac{43}{72}=0.60$	$\frac{1.5}{11}=0.14$	$\frac{7.8}{12}=0.65$
48	48	$\frac{5.5}{44}=0.13$	$\frac{47}{84}=0.56$	$\frac{3.4}{24}=0.14$	$\frac{23}{39}=0.59$	$\frac{0.86}{6.4}=0.14$	$\frac{4.4}{7.1}=0.62$
0	96	$\frac{6}{48}=0.13$	$\frac{48}{87}=0.55$	$\frac{3.6}{25}=0.14$	$\frac{21}{36}=0.58$	$\frac{1.3}{9.2}=0.14$	$\frac{2.4}{4.2}=0.57$

Isotope ^{239}Pu generates more secondary neutrons per fission reaction ν_f than ^{233}U does. However, ^{233}U generates more secondary neutrons per neutron absorption ν_{eff} than ^{239}Pu does. This is a direct consequence of smaller α values of ^{233}U compared with those of ^{239}Pu. As it follows from Table 1, the neutron multiplying properties of ^{233}U appear preferable to ^{239}Pu even for coolant density $\gamma_{H2O} = 0.1\mathrm{g/cm^3}$. This fact indicates the creation of rather softened neutron spectrum in the upper part of the SCLWR core.

Based on these results, it might be expected that fuel composition (4% ^{233}U+96% ^{232}Th) would be preferable over composition (4% ^{233}U+^{238}U-Th mixture) or (4% ^{233}U+96% ^{238}U). In order to confirm or disprove these expectations, energy dependencies of neutron absorption cross-sections were considered for ^{232}Th and ^{238}U. These dependencies are presented in Fig.2 within the energy range from 1 eV to 1 keV.

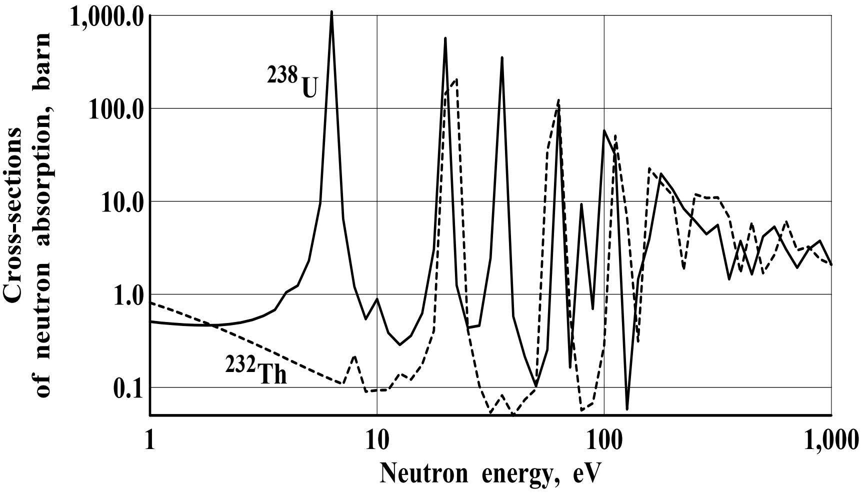
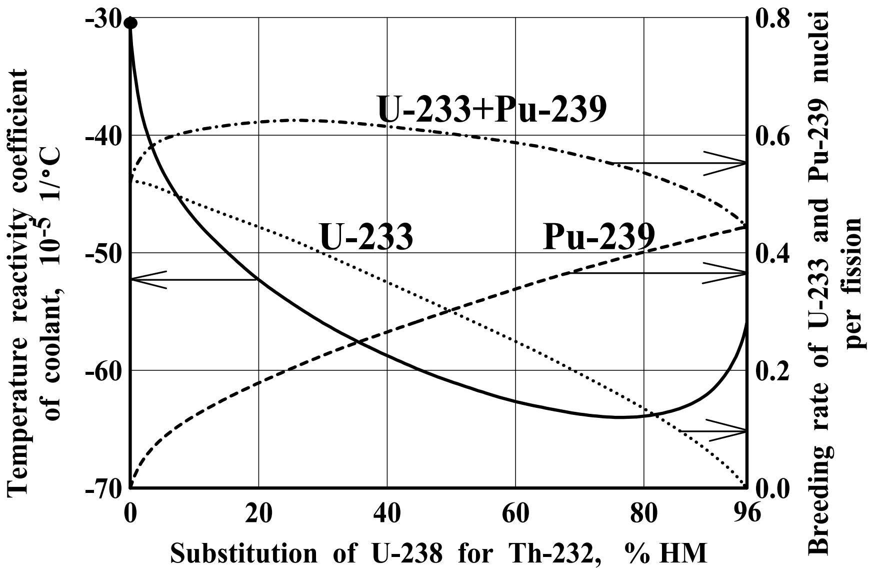


Fig.2. Micro cross-sections of neutron absorption by ^{232}Th and ^{238}U as functions of neutron energy

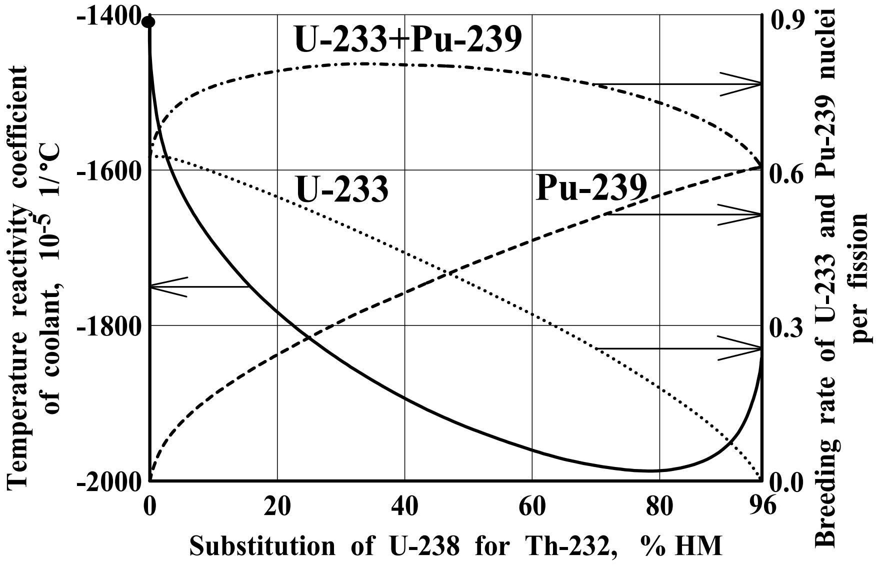
It can be seen that some neutron absorption resonances of ^{232}Th and ^{238}U do not overlap each other. That is why the resonance lattice in the case of thorium-uranium fuel composition will be denser than the lattices for ^{232}Th-based or ^{238}U-based fuel compositions. The denser resonance lattice means the larger neutron amount, in the course of neutron deceleration, will be absorbed by fertile isotopes (^{232}Th and ^{238}U) with an accordingly more intense breeding of fissionable isotopes (^{233}U and ^{239}Pu).

4. Comparison of fuel compositions on their breeding properties

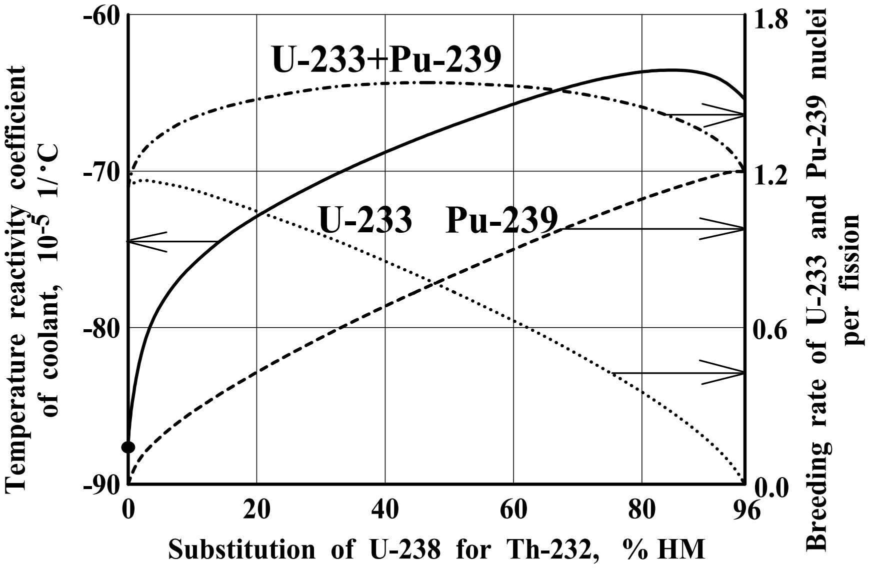
Dependencies of ^{233}U and ^{239}Pu breeding rate per fission reaction on fraction of ^{238}U are presented in Fig.3. Dioxide fuel composition (4% ^{233}U, 96% [^{232}Th+^{238}U])O$_2$ was considered for 3 different water densities. Curve of total (^{233}U+^{239}Pu) breeding rate of fissionable isotopes per fission reaction has a dome-shaped form, and maximal values of the breeding rate were obtained in (^{232}Th-^{238}U) fuel composition.



a) γ_{H2O} = 0,72 g/cm^3



b) γ_{H2O} = 0,4 g/cm^3



c) γ_{H2O} = 0,1 g/cm^3

Fig.3. Dependence of TRC and breeding rate of ^{233}U and ^{239}Pu nuclei (per fission) on fraction of ^{238}U

Total breeding ratio of ^{233}U and ^{239}Pu may be determined using the following formula that was derived from neutron balance equation:

$$KB_{U-233+Pu-239} = \frac{\nu_{eff}^{U-233}}{K_\infty} - 1 + \Delta KB_f - \Delta KB_{Structure\ +H2O} \qquad (2)$$

where: $$\Delta KB_f = \frac{\left(\frac{1}{k_\infty}\cdot\nu_f - 1\right)\cdot\Sigma_f^{Th+U-238}\int\phi\cdot dV\cdot dE\cdot d\Omega}{\Sigma_{cf}^{U-233}\int\phi\cdot dV\cdot dE\cdot d\Omega} \qquad (3)$$

$$\Delta KB_{Structure\ +H2O} = \frac{\Sigma_c^{Zr+H+FP}\int\phi\cdot dV\cdot dE\cdot d\Omega}{\Sigma_{cf}^{U-233}\int\phi\cdot dV\cdot dE\cdot d\Omega} \qquad (4)$$

Under fixed content of ^{233}U in fuel compositions, superior breeding properties of mixed ^{233}U+(^{232}Th-^{238}U) fuel composition over ^{232}Th-based or ^{238}U-based fuel compositions can be mainly explained by the dependence of neutron multiplication factor K_∞ on the fuel composition. This dependence is shown in Fig.4 for water density γ_{H2O} = 0,72 g/cm^3. The lower values of neutron multiplication factor in the case of mixed fuel composition mean larger values of the first term in the right part of formula (2), i.e. more neutrons are captured by fertile isotopes and, thus, the breeding ratio increases.

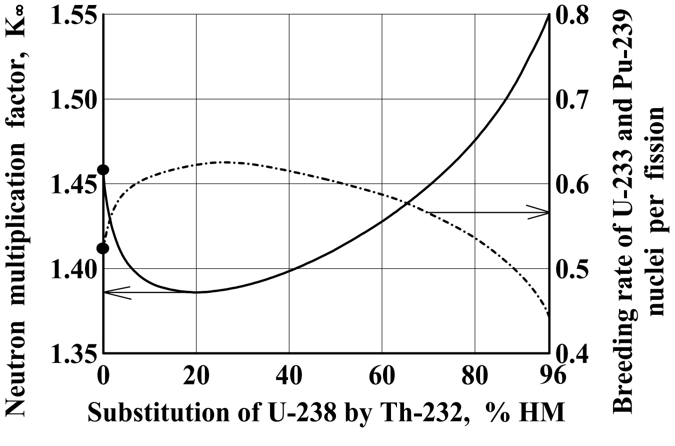


Fig.4. Dependence of K_∞ and breeding rate of ^{233}U and ^{239}Pu nuclei per fission on fraction of ^{238}U (γ_{H2O} = 0.72g/cm^3)

Furthermore, the calculations revealed that the fraction of neutrons captured by structural materials and coolant in the case of mixed fuel composition was lower than that for ^{232}Th-based or ^{238}U-based fuel compositions, and this effect was mainly caused by a denser resonance lattice in the mixed fuel. This explains the larger values of total (^{233}U+^{239}Pu) breeding ratio in mixed fuel composition.

It can be concluded from the calculated dependencies of ^{233}U and ^{239}Pu breeding rate per fission reaction on fraction of ^{238}U for two other coolant densities (γ_{H2O} = 0.4g/cm^3, γ_{H2O} = 0.1g/cm^3) that for each water density there is a certain ^{232}Th-based and ^{238}U-based fuel composition, which is able to provide better breeding properties than pure thorium or pure uranium fuel. In the case with γ_{H2O} = 0.1g/cm^3 the following fuel composition was considered: 10% ^{233}U+90% [^{232}Th+^{238}U]O$_2$. Content of ^{233}U in this fuel composition was increased to obtain approximately the same value of neutron multiplication factor K_∞ as those obtained for higher water densities.

Dependencies of temperature reactivity coefficient (TRC) on ^{238}U fraction in fuel composition are presented in Fig.3. It can be seen that TRC is negative for all fuel compositions and water densities under consideration. This provides favorable conditions for the inherent operation safety of SCLWR. The sharpest variations of coolant density occur in the central part of the reactor core. That is why TRC values are so high in the modulus for coolant density γ_{H2O} = 0.4g/cm^3.

As mentioned above, the main difference in neutron multiplying properties of ^{233}U and ^{239}Pu manifests itself in different values of the number of secondary fission neutrons per neutron absorption, i.e. in values of ν_{eff}. So, simple adding together the ^{233}U and ^{239}Pu breeding rates is not quite correct because these isotopes give remarkably different contributions to maintenance of the reactor criticality. In order to take the difference into account, a breeding rate of ^{233}U nuclei per fission reaction should be multiplied by ν_{eff}(^{233}U), the breeding rate of ^{239}Pu nuclei per fission reaction - by ν_{eff}(^{239}Pu) with further addition of these results. The corrected dependencies of ^{233}U and ^{239}Pu breeding rates are presented in Fig.5 for water density γ_{H2O} = 0.72g/cm^3.

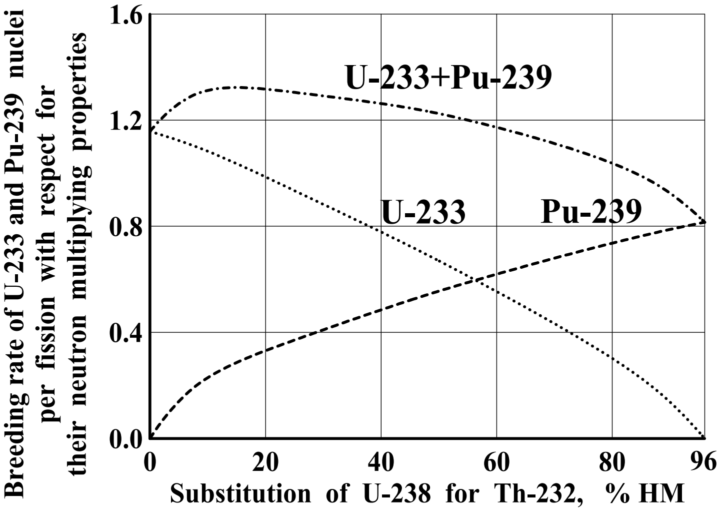


Fig.5. Dependence of ^{233}U and ^{239}Pu breeding rate per fission
with respect for different values of ν_{eff} on fraction of ^{238}U (γ_{H2O} = 0.72g/cm^3)

As it follows from Fig.5, the corrected dependence of total breeding rate of ^{233}U and ^{239}Pu nuclei per fission reaction on fraction of ^{238}U also has a dome-shaped form. However, the maximal point of the curve is shifted towards lower contents of ^{238}U (i.e. towards larger contents of ^{232}Th) in the fuel composition. Such a shift reflects the fact that neutron multiplying properties of ^{233}U are superior to those of ^{239}Pu. A similar situation was observed for cases with lower water density (γ_{H2O} = 0.4g/cm^3, γ_{H2O} = 0.1g/cm^3), i.e. (^{232}Th-^{238}U)-based fuel composition looks preferable from a fuel-breeding point of view. Incidentally, fuel composition with a 3:1 ratio of ^{232}Th content to ^{238}U content is preferable from a neutron-multiplication point of view.

5. Comparison of fuel compositions on time dependencies of neutron multiplication factor K_∞

Calculations were carried out to study time dependencies of infinite neutron multiplication factor K_∞ in the course of fuel burn-up for different fuel compositions. Three fuel compositions were considered in numerical studies, namely ^{232}Th-based fuel, ^{238}U-based fuel and (^{232}Th-^{238}U)-based fuel. Contents of fissionable isotope ^{233}U in each fuel composition were chosen so that the initial values of K_∞ for all fuel compositions were identical. In the case of (^{232}Th-^{238}U)-based fuel, the ratio between contents of these isotopes is equal to 3 (the optimal ratio for reaching maximal fuel burn-up).

Time dependencies of K_∞ for all three fuel compositions and for water density γ_{H2O} = 0.72g/cm^3 are presented in Fig.6. As is evident, K_∞ of ^{238}U-based fuel falls with the greatest speed. In fact, due to better neutron multiplying properties of ^{233}U compared with ^{239}Pu, K_∞ of ^{232}Th-based fuel does not fall as quickly. Fuel based on (^{232}Th-^{238}U) mixture looks the most attractive because achievable fuel burn-up of this fuel composition is higher by a factor of 1.3 compared with that for ^{232}Th-based fuel and by a factor of 1.75 over that for ^{238}U-based fuel.

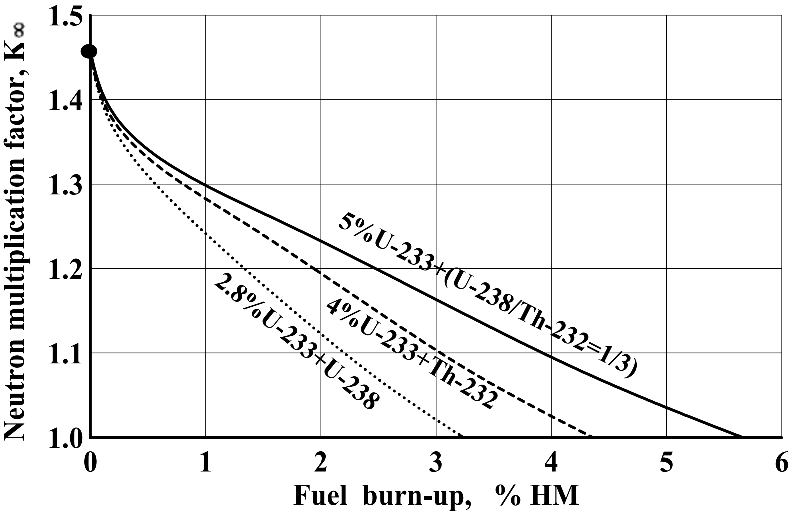


Fig.6. Dependence of K_∞ on fuel burn-up (γ_{H2O} = 0,72 g/cm^3)

In cases of lower water densities (γ_{H2O} = 0.4g/cm^3, γ_{H2O} = 0.1g/cm^3), the behavior of K_∞ time dependencies essentially does not change. So, the curves confirm the attractiveness of (^{232}Th-^{238}U)-based fuel composition because of the highest fuel burn-up in comparison with ^{232}Th-based and ^{238}U-based fuel compositions.

6. Use of stainless steel as a cladding material for fuel rods and assemblies

Numerical studies on neutron-physical parameters of SCLWR, as presented above, were carried out under the assumption that zirconium-niobium alloy, traditional for LWR, is also used in SCLWR as a cladding material. However, as it was noted in Ref. 2, that high coolant temperature makes it impossible to use zirconium-based alloys and "requires the use of alloys with a high nickel content that substantially worsens the parameters of the nuclear fuel cycle". Therefore, transition from zirconium to stainless steel cladding and induced changes of neutron-physical parameters constituted the subject of a separate study for the proposed uranium-thorium fuel compositions of SCLWR.

Neutron spectrum-averaged micro cross-sections of parasitic neutron absorption by coolant and structural materials were determined using the SCALE-4.3 computer code package [5]. These cross-sections for three water densities (γ_{H2O} = 0.72g/cm^3, γ_{H2O} = 0.4g/cm^3, γ_{H2O} = 01g/cm^3) and three fuel compositions are presented in Table 2 (cladding material - Zr+1%Nb alloy) and in Table 3 (cladding material - stainless steel containing iron, chromium and nickel). The following three fuel compositions were analyzed: ^{232}Th-based fuel composition, i.e. 4% ^{233}U+96% ^{232}Th; ^{238}U-based fuel composition and (^{232}Th-^{238}U)-based fuel composition (ratio of ^{232}Th content to ^{238}U content is equal to 3). For all three values of water density, the content of fissionable isotope ^{233}U in ^{238}U-based and (^{232}Th-^{238}U)-based fuel compositions was chosen so that the initial values of K_∞ were equal to the initial K_∞ of the ^{232}Th-based fuel composition. As in the previous study, a 10%^{233}U+90%^{232}Th fuel composition was considered for the case of the least water density (γ_{H2O} = 0.1g/cm^3). So, an increased content of ^{233}U was chosen to obtain approximately the same values of K_∞ as in cases of larger water densities.

Table 2

Neutron absorption cross-sections of coolant (H_2O) and cladding material (Zr+1%Nb)

Fuel	$\gamma_{H2O} = 0.72 g/cm^3$			$\gamma_{H2O} = 0.4 g/cm^3$			$\gamma_{H2O} = 0.1 g/cm^3$		
	H_2	Zr	Nb	H_2	Zr	Nb	H_2	Zr	Nb
^{232}Th	0.026	0.043	0.40	0.014	0.040	0.40	0.0012	0.035	0.36
^{238}U	0.036	0.047	0.42	0.018	0.042	0.40	0.0013	0.035	0.37
^{232}Th+^{238}U (3:1)	0.022	0.042	0.39	0.010	0.039	0.39	0.0009	0.034	0.34

Table 3

Neutron absorption cross-sections of coolant (H_2O) and cladding material (stainless steel)

Fuel	$\gamma_{H2O} = 0.72 g/cm^3$				$\gamma_{H2O} = 0.4 g/cm^3$				$\gamma_{H2O} = 0.1 g/cm^3$			
	H_2	Cr	Fe	Ni	H_2	Cr	Fe	Ni	H_2	Cr	Fe	Ni
^{232}Th	0.024	0.22	0.18	0.32	0.013	0.13	0.11	0.18	0.0012	0.028	0.020	0.030
^{238}U	0.029	0.26	0.22	0.38	0.015	0.15	0.12	0.20	0.0013	0.029	0.021	0.031
^{232}Th+^{238}U (3:1)	0.021	0.19	0.16	0.27	0.010	0.10	0.08	0.14	0.0009	0.026	0.018	0.026

As is evident from Tables 2 and 3, for all three water densities, mixed (^{232}Th-^{238}U)-based fuel composition provides lower values of parasitic neutron absorption cross-sections both by coolant and cladding materials when compared with the two other fuel compositions. One of the causes for this effect is related with a denser lattice of neutron absorption resonances. This conclusion is correct for both structural materials (Zr+1%Nb alloy and stainless steel).

Zirconium-based alloys are used as a cladding material for fuel rods in light-water VVER-type reactors because of low parasitic neutron absorption. In this connection, it is noteworthy that, in the case of the least water density ($\gamma_{H2O} = 0.1 g/cm^3$) and mixed (^{232}Th-^{238}U)-based fuel composition, spectrum-averaged cross-sections of neutron absorption in main components of stainless steel appeared lower than those for the main components of the zirconium-based alloy.

Fission products are parasitic neutron absorbers, too. However, it is very difficult to compare spectrum-averaged micro cross-sections of parasitic neutron absorption by each fission product. Therefore, computations were carried out to determine the spectrum-averaged micro cross-sections of parasitic neutron absorption by a single "effective" fission product of ^{235}U and ^{239}Pu, respectively, with application of the GETERA computer code [6]. Thus, it was assumed that neutron absorbing properties of fission products generated by ^{233}U are identical to those generated by ^{235}U. Neutron-spectrum-averaged micro cross-sections of parasitic neutron absorption by "effective" fission products of ^{235}U and ^{239}Pu, respectively, are presented in Table 4 for all three fuel compositions, for both cladding materials and for the largest value of water density ($\gamma_{H2O} = 0.72 g/cm^3$).

Table 4

Neutron absorption cross-sections (barns) by "effective" fission products of ^{235}U and ^{239}Pu
at the beginning of fuel campaign

Fuel composition	Cladding material for fuel rods and assemblies			
	Zr+1%Nb		Stainless steel	
	^{235}U	^{239}Pu	^{235}U	^{239}Pu
^{232}Th-based	7.5	9.6	7.3	9.4
^{238}U-based	8.5	10.6	8.0	10.1
(^{232}Th+^{238}U)-based	6.6	8.5	6.4	8.3

As for neutron absorption by fission products, the following conclusions can be derived from Table 4. The fuel composition based on (^{232}Th-^{238}U)-mixture is preferable over the fuel compositions based on ^{232}Th and on ^{238}U, respectively, for both cladding materials. Moreover, stainless steel is preferable to zirconium-based alloy from the same point of view.

The SCALE-4.3 computer code package was used to determine achievable values of fuel burn-up. The results obtained in these computations are presented in Table 5 for the case of stainless steel as a cladding material.

Table 5

Achievable fuel burn-up (% HM)

Fuel composition	γ_{H2O} = 0.72g/cm^3	γ_{H2O} = 0.4g/cm^3	γ_{H2O} = 0.1g/cm^3
^{232}Th	3.02	2.68	11.47
^{238}U	2.70	2.51	7.73
^{232}Th-^{238}U	4.37	3.8	13.91

Transition from zirconium-based alloy to stainless steel as a cladding material did not change the overall picture because the better neutron multiplying properties of ^{233}U compared with ^{239}Pu resulted in higher values of achievable fuel burn-up in ^{232}Th-based fuel than those in ^{238}U-based fuel. Nevertheless, a mixed (^{232}Th-^{238}U)-based fuel composition looks the most attractive option thanks to lower cross-sections of neutron absorption in coolant and structural materials, and higher values of effective neutron breeding ratios with proper regard for the difference in neutron multiplying properties of produced fissionable isotopes.

7. Conclusion

Neutron-physical properties of (^{233}U-^{232}Th-^{238}U) fuel compositions were studied under the operating conditions of a power light-water reactor with super-critical parameters of coolant (SCLWR).

Within the SCLWR operation range of water density (from 0.78g/cm^3 to 0.09g/cm^3), ^{233}U remains remarkably more preferable than ^{239}Pu on neutron multiplying properties: $\nu_{eff}(^{233}U) = 2.22$; $\nu_{eff}(^{239}Pu) = 1.82$.

It is demonstrated that mixed (^{232}Th-^{238}U)-based fuel composition provides higher values of achievable fuel burn-up than ^{232}Th-based and ^{238}U-based fuel compositions, respectively. In addition, a three-to-one ratio of ^{232}Th and ^{238}U contents in mixed (^{232}Th-^{238}U)-based fuel composition provides maximal values of fuel breeding ratio and achievable fuel burn-up.

It is demonstrated that cross-sections of parasitic neutron absorption by coolant, structural materials and fission products are lower in mixed (^{232}Th-^{238}U)-based fuel composition than those in ^{232}Th-based and ^{238}U-based fuel compositions, respectively. Such a conclusion is correct for both cases of cladding materials: zirconium-based alloy (Zr+1%Nb) and stainless steel.

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