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Journal of Nuclear Materials 319 (2003) 154–158

journal of nuclear materials

www.elsevier.com/locate/jnucmat

The feasibility of employing inert matrix ceramic fuels in a Russian light water reactor

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Abstract

This study presents a version of Water–Water Energy Reactor (WWER) type fuel elements made using inert matrix fuels (IMFs) whose oxide fissile particulates are densely placed throughout a highly heat-conductive metal matrix consisting of silumin (cermet fuels). In these WWER-type fuel elements with cermet fuel, the fuel is bonded to cladding in a metallurgic way. It possesses some special features allowing the fuel to be used for both steady-state and load following operation modes. The study presents the results of the analytic study of the two variants of cermet fuel based on UO_2 and $UO_2 + ThO_2$. The use of burnable poisons Er_2O_3 and Gd_2O_3 was considered. Characteristics of WWER with $UO_2 + ThO_2$ cermet fuel has been compared with those for conventional ceramic uranium dioxide fuel. In-pile testing of UO_2 cermet fuel elements was carried out successfully up to a burnup of $\cong$60 MW d kg^{-1} of U. Results of stress calculations have shown fuel element cladding stability for operation in the load following mode.

1. Introduction

Cermet fuel is a variety of composite inert matrix fuel (IMF). It consists of ceramic oxide fissile particulates surrounded by a metal matrix possessing good physical and thermo-physical characteristics. As a rule, the cermet fuel is metallurgically bonded to the cladding. A cermet fuel element is a monolithic structure.

This study presents two versions of the fuel of the forenamed kind containing a silumin (aluminium-based alloy) matrix. The cermet fuel is bonded to the cladding made of a zirconium-based alloy (Zr + 1%Nb). The volume fraction of the ceramic fissile component is present to a maximum of 60 vol.%, so the matrix material fraction is present in an amount of more than 40 vol.%.

Thanks to a high thermal conductivity of silumin and to a negligible thermal resistance of the cermet-cladding bond interface, the temperature drop between the cladding and the centre of the fuel pin becomes considerably less in comparison with the standard design WWER (Water–Water Energetic Reactor) fuel elements.

The information related to such WWER cermet fuel elements has been presented in [1,2].

Two aspects motivating the application of cermet fuel in the WWER-type reactors are examined in this paper:

(a) the assurance of the Nuclear Power Plant (NPP) capability to be operated in a load following mode;
(b) the feasibility of reducing the stock of Pu and minor actinides by means of WWERs.

Introduction of cermet fuels into the WWERs will not require changes of both the fuel assembly (FA) design and the fuel element dimensions. Such an approach severely requires the obligatory conservation of the fuel assembly power and, consequently, the ^{235}U load both per individual fuel element and per individual fuel assembly.

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0022-3115/03/$ - see front matter © 2003 Elsevier Science B.V. All rights reserved.
doi:10.1016/S0022-3115(03)00211-3

2. The feasibility of assuring the load following mode operation of a nuclear power plant based on the WWER with a cermet fuel core

2.1. Neutronic calculations referring to WWER loaded with cermet fuel

Since the uranium content in a cermet fuel is reduced (by ≈29%) as compared to that in the uranium dioxide case, it results a reduction of the refuelling periods. Providing the uranium enrichment is kept lower than 5%, the reactor refuelling period will be of 255 effective days, which can be recognised satisfactory for a reactor designed to operate in the load following mode.

The maximum fuel burnup in the FA is of 58 MW d kg^{-1} of U.

The reactivity coefficients of the core during operation at a nominal power level meet the regulatory requirements with respect of sign and value.

In order to ensure a negative temperature coefficient of reactivity at a temperature corresponding to a minimal controllable power level (MCPL), currently characterized by $T_{H_2O} = 553$ K at the beginning of core life, the use of burnable poisons has been examined. They can presently reduce the initial concentration of boron acid. The application of burnable poisons has been examined for the variants of fuel elements containing gadolinium and those containing erbium (in the form of the so called integrated uranium–erbium fuel, UEF). In case of the uranium–erbium fuel, apart from the reduction of boron concentration in the coolant, erbium yields also an additional negative component of the temperature coefficient of reactivity.

The principal results of the calculations of fuel cycle characteristics are given in Table 1. A 3D diffusion 2-group code was used for calculations, which allows to model thermo-hydraulic feedback. Group cross-sections and diffusion coefficients were calculated using the WIMS-D4 code.

2.2. Computational and experimental study of cermet fuel element performance in the load following mode

The analysis of the presented computational data referring to the design of a cermet fuel element shows that the maximum temperatures at the fuel centre are:

Table 1
Characteristics of WWER-1000 related to two types of burnable neutron poisons

Characteristic	The way of employing a burnable neutron poison with	
	12 Gd-containing fuel elements per FA	U–Er fuel in all elements
^{235}U enrichment in fuel elements/%	4.0	–
Average enrichment in the replenishing fuel[a]/%	4.66	4.69
Load of uranium per a fuel assembly/kg	307.8	307.6
Reactor refueling interval, effective days/d	254.55	252.76
Value of k_q in the beginning of the refueling interval[b]	1.40	1.44
Value of k_q in the end of the refueling interval	1.32	1.30
Value of k_v in the beginning of the refueling interval[b]	1.72	1.75
Value of k_v in the end of the refueling interval	1.48	1.44
Height-averaged burnup of fuel in unloaded Fas/MW d kg^{-1}:		
• Average	51.3	51.0
• Maximum	54.2	53.9
Maximum burnup of fuel in fuel assemblies/MW d kg^{-1}	57.6	57.6
Critical concentration of boron in the coolant/ppm	1202	1243
The concentration of boron at a nominal level of power[c]/ppm	819	851
Temperature coefficient of reactivity (by the coolant temperature) at a minimally controllable power level (MCPL) in the load following mode (at $T_{H_2O} = 553$ K)/10^{-5} K^{-1}	−1.52	−3.90
The same as above at a nominal level of power/10^{-5} K^{-1}	−20.7	−24.2
Temperature at which the value of $(\partial\rho/\partial T_{H_2O})$ at a MCPL in the beginning of the core life-time becomes negative/K	541	516

[a] Regarding the central FA.
[b] After the reactor poisoning (in 2 effective days). k_q – FA power picking factor, k_v – power density picking factor.
[c] Disregarding a FA power picking factor.

$T_{fuel}^{max} \cong 718$ K in the WWER-440 reactor;

$T_{fuel}^{max} \cong 783$ K in the WWER-1000 reactor.

At these low temperatures the fission products remain located in dioxide fissile particulates of the cermet fuel and in the close regions around them.

In addition, the thermal diffusion of the fission products is impeded at a level that can be neglected. In this case, fission products will penetrate to certain depths in the matrix material only at the expense of the energies acquired by fission. Thus in the course of a fuel element operation a spherical definite radius zone of the material damaged by matrix fission products will be formed eventually around each fissile particulate. With a fissile particulate diameter being of 420 μm, only a small fraction (0.034) of the fission products formed within the particulate will penetrate into the matrix. The disturbance of the cladding in the stationary operation mode takes place with its specified deformation rate at the expense of the fuel composition. The contact of the cladding with corrosion-aggressive fission products is totally absent. So the danger of the stress corrosion cracking (SCC) of a cermet fuel element cladding is also completely absent.

The working efficiency of the cermet fuel element cladding operating in stationary conditions is determined by its in-reactor deformability. The (Zr + 1%Nb) alloy employed in WWER reactors as a fuel element cladding material possesses a very high deformability under irradiation. When a fuel element cladding has been subjected to the in-reactor deformation at a rate of $\cong 10^{-4}$% h^{-1}, its failure has not been observed at strains up to ε_i, of 7%.

The computational data show that at a burnup (B) of 60 MW d kg^{-1} of U the strain value ε_i will be $\cong$ 2.0%, which is considerably lower than the strain to failure (breaking). The change of fuel element cladding diameter, Δd, at $B = 60$ MW d kg^{-1} of U, equals $\sim$+0.15 mm ($\varepsilon_i = 1.70$%).

The working efficiency of the cermet fuel element cladding when operating in the load following mode must be determined by its cyclic strength (provided that danger of the stress corrosion cracking is absent). The computation has shown also that the change in the reactor power down to the shut-down cooling results in the rise of thermo-mechanical stresses in the fuel element cladding with an amplitude, $\Delta\sigma_a$, of $\cong$120 MPa. This is a cyclic loading with a stress amplitude being lower than double the yield stress of the cladding material, the (Zr + 1%Nb) alloy: $\Delta\sigma_a < 2\sigma_T$. In this case substantial cyclic damages of the cladding material can take place only at a considerable ($N \geqslant 10^6$) number of cycles from 100% to 0% of nominal power (P_{nom}) and back. Therefore, load following operation with power change in the range of 0–100% of P_{nom} is acceptable if overall number of cycles in less than 10^6.

The computational data has shown also that in a varying operation mode a cladding remains always pressed against a fuel core ($\sigma_r < 0$). Thus the cause of stripping the cladding off the cermet fuel in the course of operation is absent.

The service life testing of the cermet fuel elements carried out in the Research Institute of Atomic Reactors (RIAR, Dimitrovgrad) in the Materials Investigation Reactor (MIR) has shown that the fuel elements have functioned up to the maximum burnup, B_{max}, of $\cong$60 MW d kg^{-1} of U without any cladding failures and successfully endured the changes in the power of the reactor in the course of its operation.

3. The employment of the urania–thoria cermet fuel in the WWER-type reactors

The concept of introducing the urania–thoria cermet fuel in the WWER-type reactors is based upon preserving the current design of the fuel assembly and that of the reactor core and presumes upon the feasibility of a gradual conversion to this new kind of fuel. In this case FA power and uranium-235 loading has to be kept inalterable. At present the load of an individual WWER-1000 fuel element (whose active length, H_A, is 355 mm high) is 1460 g UO_2, with the uranium enrichment of 4.4%. In our case, with 20% enrichment, it will be equivalent to $\approx$320 g UO_2. Proceeding from the condition of preserving the reactor life-time, the remaining fraction of the space that is available inside the cladding is partially filled with ThO_2 (with it both the cladding material and the cladding dimensions, $\varnothing 9.1 \times 0.69$ mm, remained unchanged). The filling must be only partial as ^{232}Th has a thermal neutron absorption cross-section almost three times as high as that of ^{238}U. The determination of the necessary amount of thorium in a fuel element becomes one of the goals of the neutron physics computation. The remaining fraction of the space inside a fuel element is to be filled with silumin as a well thermal-conducting matrix material.

Apart from the initiation of introducing the uranium–thorium fuel cycle into the nuclear power unit, the proposed approach would allow to realize the specific tasks of improving the safety and economical efficiency of the WWER-1000 reactors since it results in the reduction of the ^{135}Xe poisoning, of the excess reactivity for a core life-time, and, of the temperature level of these fuel elements.

The characteristics of the uranium–thorium cermet fuel assumed for these computations are presented in Table 2 in comparison with those of a uranium dioxide pellet fuel.

Table 2
Comparative characteristics of WWER-1000 fuel elements with urania and urania–thoria fuels

	Fuel			
	UO_2		$UO_2 + ThO_2$	
Enrichment of uranium/%	4.4	3.6	20	16
Number of fuel elements in profiled fuel assemblies/–	240	72	240	72
Mass of UO_2 per fuel element/g	1460	1460	320	320
Mass of ^{235}U per fuel element/g	56.0	45.8	56.0	44.8
Mass of ThO_2 per fuel element/g	–	–	650	650
Volume fraction of fuel in pellets/–	1.00	1.00	0.58	0.58
Volume fraction of the filler (Si–Al alloy)/–	–	–	0.42	0.42

Table 3
Burnup fractions of fuel in the unloaded fuel assemblies

	Value	Heavy atom burnup/ $MW\,d\,kg^{-1}$	Fission product fraction/ $g\,cm^{-3}$
Average along the height of a fuel assembly	Minimum	56.6	0.323
	Mean	62.3	0.355
	Maximum	73.7	0.420
The maximal value for a layer of $\Delta Z = 1/20H_{core}$		79.1	0.451

Table 4
The amounts of minor actinides built-up in the WWER-1000/ $kg\,a^{-1}$

Isotope	UO_2 (with 4.4% ^{235}U)	UO_2–ThO_2–silumin (with 20% ^{235}U)
^{237}Np	12.50	10.92
^{238}Np	0.0407	0.0419
^{239}Np	2.343	0.858
^{241}Am	0.743	0.230
^{242}Am	0.0140	0.0037
^{243}Am	2.857	1.45
^{242}Cm	0.370	0.152
^{243}Cm	0.0106	0.00430
^{244}Cm	0.740	0.390
^{245}Cm	0.035	0.0166

Table 5
Characteristics of the WWER-1000 reactor with urania–thoria cermet fuel and UO_2 pellet fuel

	Fissile composition			
	UO_2		UO_2 + ThO_2 + silumin	
Mass of uranium per fuel assembly/kg	401		88	
Mass of thorium per fuel assembly/kg	–		178	
Phase of the life-time	Beginning	End	Beginning	End
Fuel assembly power picking factor	1.27	1.23	1.29	1.22
Temperature coefficient of reactivity at a MCPL (at $T_{H_2O} = 553$ K) $10^{-5}/K^{-1}$	−3.61	−24.8	−3.56	−21.7
Efficiency of the emergency protection system at a nominal power/%	8.39	7.80	7.63	7.84
Critical concentration of boron in the coolant at a MCPL (at $T_{H_2O} = 553$ K)/$g\,kg^{-1}$	1.74	0.69	1.30	0.41
Temperature of recurring criticality/K	–	433	–	319
Annular consumption of $^{235}U/t\,a^{-1}$	0.921		0.949	
Annular consumption of thorium/$t\,a^{-1}$	–		9.61	
Content of ^{232}U in discharged uranium/ppm	–		155	
Annular buildup of plutonium/$kg\,a^{-1}$	245		80	
Annular buildup of $^{233}U/kg\,a^{-1}$	–		173	

In order to equalize the values of the fuel element power, the fuel assemblies have been profiled by the uranium enrichment. The reactor-recharging ratio (the ratio of the total amount of FAs in the core to the amount of the FAs being recharged) equals to three. The reactor life-time is 292 effective days, and the life-time of a FA is of three calendar years.

The burnup fractions of fuel in the unloaded fuel assemblies are presented in Table 3. It varies of ~40% along the height of a fuel assembly.

The data relating to the buildup of minor actinides in the fuels are given in Table 4. ^{237}Np, ^{243}Am and ^{244}Cm are the predominant minor actinides in the spent fuel.

Some other characteristics of the WWER-1000 with the urania–thoria cermet fuel in comparison with those with the conventional urania-based fuel are presented in Table 5.

4. Conclusions

Cermet fuel based on UO_2 in silumin inert matrix seems to be promising first of all for using in the WWER operating in a load following mode. This fuel is characteristic of a low cladding stress in a cyclic reactor power change during operation. The positive results of in-pile tests of this type of fuel are available.

Both UO_2 cermet and UO_2–ThO_2 cermet fuels are applicable to WWER. Uranium enrichment does not exceed 20% in this case. Conventional UO_2 fuel could be step by step replaced by mixed UO_2–ThO_2 cermet fuel for one of the operating WWER.

Using the UO_2–ThO_2 cermet fuel for WWER-1000 can be characterized by the following advantages:

- lower (about three times) buildup of plutonium and several minor actinides,
- reduced (by 23%) poisoning of the reactor by ^{135}Xe that will contribute to the prevention of cyclic power fluctuations,
- improvement of the reactor plant safety at the expense of reducing the critical concentration of boron in coolant in the beginning of refueling interval from 1.07 to 0.85 $g\,kg^{-1}$ of H_2O.

Study on UO_2–ThO_2 cermet fuel application is now at an analytical and initial fuel technology development stage.

References

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