Concept of Multifunctional Fast Neutron Research Reactor (MBIR) Core with Metal (U-Pu-Zr)-fuel

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Abstract. Multifunctional fast neutron research reactor MBIR is intended to provide the basis for broad scope of research and experimental activities. The focus of this work is the analysis of different MBIR fuel types (vibrio-packed MOX-fuel and metal fuel) to provide required neutron flux. On the basis of 5.5 mm U-Pu-10Zr fuel pin with sodium sub-layer efficient MBIR core can be introduced to meet all requirements and expand consumer needs for this reactor as experimental facility.

Key Words: Multifunctional fast neutron research reactor MBIR; metal fuel; sodium sub-layer; neutron flux.

1. Introduction

Multifunctional fast neutron research reactor MBIR is intended to provide the basis for broad scope of research and experimental activities including experiments with advanced fuel types, absorbing and structural materials and coolants and research of issues of nuclear fuel cycle closing and radioisotopes production. Moreover MBIR reactor assumes broad scope of applied and medical research based on neutron beams. Fast power reactor under operation (e.g. BN-600) is not very suitable facility for these purposes because of licensing issues. This is one of the main reasons to use research reactor for such activities. Present day Russia has BOR-60 research reactor under operation, however its lifetime is close to the end. MBIR reactor should become new research facility.

MBIR reactor construction provides significant contribution to development of innovative experimental base. First of all, it is related to achieving of high neutron flux in experimental channels and carrying out research of fuel types and structural materials for advanced reactors with various coolants from gas to molten salt on the basis of independent circulation loops. MBIR reactor is assumed to become modern international instrument for experimental research of innovative fast reactor technologies.

One of the main requirement for MBIR reactor is achieving of 5.0·1015 cm-2•s-1 neutron flux in central loop channel. That is why MBIR reactor is considered as strong neutron source.

The aim of this work is to find reliable reserve fuel option unless basis option of vibrio-packed MOX-fuel will be ready to use. Turning to reserve fuel option should be as simple as possible.

International experience shows that countries requiring high plutonium production rate will use both MOX- and metal fuel in their power rectors on fast neutrons. Metal fuel also enables us to improve inherent safety features of fast sodium cooled reactors. This is the reason for development of metal fuel option for MBIR reactor because this fuel type has number of advantages as research reactor fuel on the one hand and this experience will be useful for development of metal fuel for power reactors on the other hand.

This work focuses on several fuel types for MBIR reactor, which helps to achieve required neutron flux in loop channels and experimental assembly positions. Three fuel options are considered in this study: 6 mm diameter fuel pin with vibrio-packed MOX-fuel, with metal (U-Pu-Zr)-fuel and gaseous sub-layer and fuel pin of decreased diameter (5.5 mm) with metal (U-Pu-Zr)-fuel and sodium sub-layer.

In the last option (with sodium sub-layer) fuel pin structure was slightly changed: as a result of gap filled with sodium, gas plenum was arranged in top part of the fuel pin. In all options considered fuel assembly contains the same amount of fuel pins. In metal fuel with sodium sub-layer option original heavy nuclide fractions were set to keep the same amount of uranium and plutonium as in the vibrio-packed MOX-fuel option. However further plutonium content was increased on 1.4 % to provide sufficient margin for reactivity drop.

The task of reserve fuel option development requires simulation of neutronic characteristics and isotopic kinetics, core thermo-hydraulic and fuel pin stress-strain behavior analysis.

2. Fuel Characteristics

Main fuel characteristics for all options considered are shown in Table I. Plutonium isotopic composition is the same for all options: Pu-238 – 0.13 %; Pu-239 – 91.72 %; Pu-240 – 6.55 %; Pu-241 – 1.17 %; Pu-242 – 0.43 % (масс.).

TABLE I: MAIN FUEL CHARACTERISTICS

Option title	MOX	MET	MET2a
Core fuel type	Vibrio-MOX	U-Pu-10Zr alloy	U-Pu-10Zr alloy
Smeared fuel density in the core, g/cm^3	9.0	15.0	10.8 (15.4 for pellet)
Pu fraction, m/o	38.5	20.9	32.4
Blanket fuel type	Depleted UO2	U-10Zr alloy	U-10Zr alloy
Smeared fuel density in the blanket, g/cm^3	9.5	15.0	11.0

In vibrio-packed MOX-fuel option core part of fuel pin consists of mixture of vibrio-packed MOX-fuel (93 m/o) and metal depleted uranium (7 m/o). U-235 fraction in depleted uranium is 0.4 %.

In the second option of high density (15 g/cm^3) metal fuel with gaseous sub-layer fuel pin structure stays the same because of direct fuel replacement.

In the third option of metal fuel (with sodium sub-layer) fuel pin structure was slightly changed: as a result of gap filled with sodium, gas plenum was arranged in top part of the fuel pin. The main technical solution and their reasons are as follows:

– fuel pin diameter decreasing to provide increased core thermo-hydraulic margins and low coolant heating in the core;

– small increasing of plutonium content (1.4 %) compared to initial vibrio-packed MOX-fuel option to provide sufficient margin for reactivity drop;

– increasing of initial gap between fuel kernel and cladding to provide required porosity (30 %) and to compensate fuel swelling until gaseous fission products release under cladding;

– the gap between fuel kernel and cladding is filled with sodium to provide fuel to cladding heat transfer;

– gas plenum in the top part of the fuel pin because of sodium sub-layer.

FIG.1 shows structure of 5.5 mm diameter fuel pin with cladding thickness of 0.3 mm and fuel kernel diameter of 4.1 mm. Initial gap fraction inside fuel pin is 30 %.

Fuel kernel diameter and heavy nuclide fractions were set to keep the same amount of uranium and plutonium as in the initial vibrio-packed MOX-fuel option. However, plutonium content was slightly increased (1.4 %) to provide sufficient margin for reactivity drop.

As a result, the third option assumes the same fuel assembly containing the same number of fuel pins of decreased diameter.

Fuel assembly life time is defined by operational limitations such as fast neutron fluence on cladding, burnup and damage dose. It is turned out that fast neutron fluence becomes the main limitation factor because it achieves its critical value ($1.5 \cdot 10^{23}$ cm^{-2}) much earlier than discharge burnup (14 %) and damage dose (90 DPA).

3. Simulation of Isotopic Kinetic and Reactor Lifetime

Fuel burnup is simulated with two groups of codes: precise ISTAR system, based on MNCP5 neutronic code with ENDF-B/VII nuclear data, and engineering codes JARFR [2] and ShIPR [3], based on solution of multi-group diffusion equation, which provides fast estimation of large number of neutronic characteristics even with spatial distribution.

This work does not assume replacement of fuel assemblies achieved limiting parameters on fluence, burnup and damage dose. In all burnup simulations fresh core burned during 5 cycles of 100 EFPD without any replacements and cooling periods.

The beginning of the third cycle (after 200 EFPD burnup) is considered as average stationary state.

Control rods positions and initial plutonium content are chosen to provide zero reactivity by the end of the third cycle. The core contains 8 control rods consisting of 2 scram rods, 2 compensators, 2 regulators and 2 automatic regulators. Scram rods and compensators are removed from the core, automatic regulators are inserted in the core on a half of its lengths and position of the rest two regulators provides criticality in the end of the third cycle.

All simulations are carried out for the reactor consisting of the core with 93 fuel assemblies, in vessel storage with 47 cells (38 are intended for irradiated fuel assemblies) and one row of shield assemblies on the periphery. Core layout is shown in FIG.2.

By means of ISTAR system reactivity drop over reactor life-time and neutron flux in central loop channel were estimated. JARFR and ShIPR codes were used to estimate reactivity drop over reactor life-time, spatial distribution of energy generation density and radiation damage production rate.

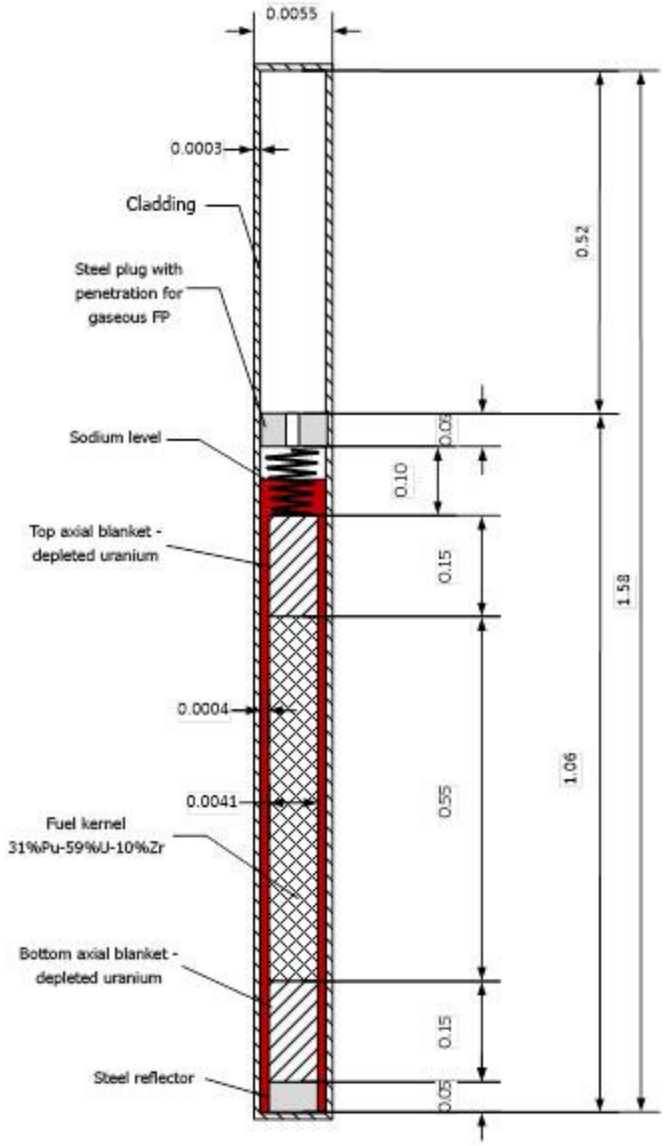

Fig. 1. U-Pu-10Zr fuel pin with sodium sub-layer structure. All sizes are in meters.

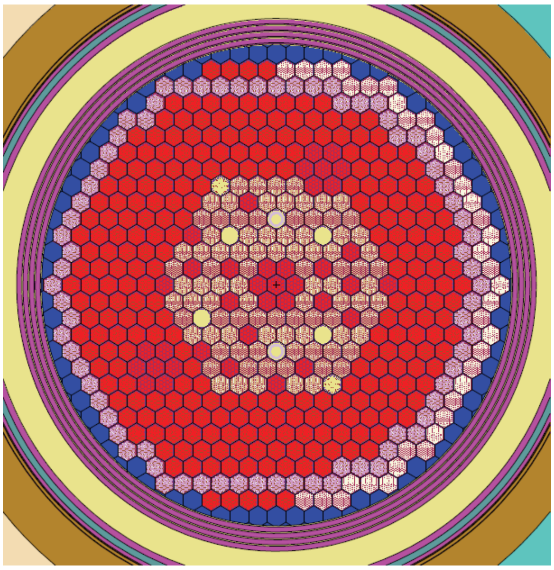

Fig. 2. MBIR core layout

Main characteristics of MBIR reactor with considered fuel types are shown in Table II. “MOX” column describes reactor characteristics for vibrio-packed MOX-fuel option, columns “MET” and “MET2a” describes metal fuel options with gaseous and sodium sub-layer respectively.

TABLE II: MAIN CHARACTERISTICS OF MBIR REACTOR WITH DIFFERENT FUEL TYPES

	MOX	MET	MET2a
Initial core Pu-239 loading, kg	277.9	275.4	281.9
Max linear heat load, kW/m	45.3	45.6	42.9
Max core burnup, % h.n.	11.3	6.7	11.1
Average core burnup, % h.n.	8.7	5.2	8.8
Reactivity drop over the 3rd cycle, %Δk/k	2.76	1.87	2.62

Maximum burnup over 500 EFPD is estimated as average burnup multiplied by radial peaking factor K_r^{max}.

FIG.3 shows reactivity drop over rector life-time for three fuel options.

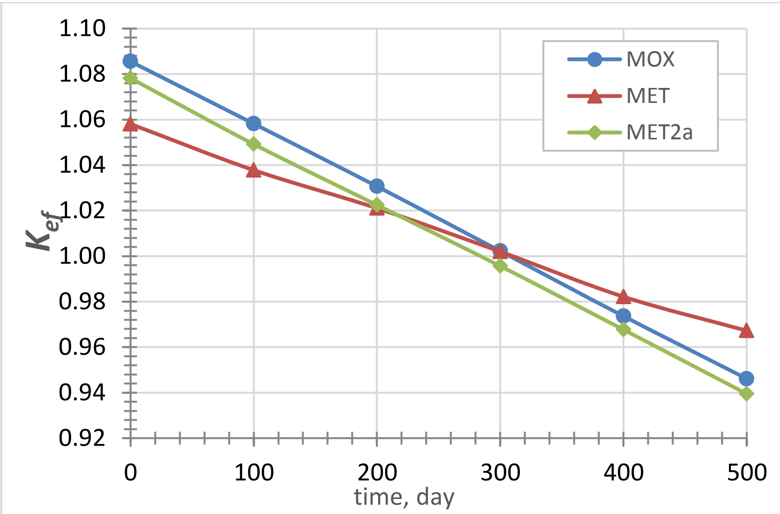

Fig. 3. Time dependency of reactivity drop: MOX – vibrio-packed MOX-fuel, MET – high density metal fuel (15 g/cm^3), MET2a – metal fuel with sodium sub-layer in 5.5 mm fuel pin.

Average neutron flux in central loop channel is estimated as neutron flux, averaged over the volume of 19 displacer rods inside central assembly. This region is highlighted on fig. 4 with green color. Matrial compositions of all displacers inside central loop channel are the same. Different colors are used only for clarity.

This region was split over 19 5 cm thick layers by height. This approach enables us to receive axial distribution of neutron flux in central loop channel. It should be noticed that there is some asymmetry in axial shape of neutron flux distribution resulted from asymmetric burnup of top and bottom blankets (because of control rods).

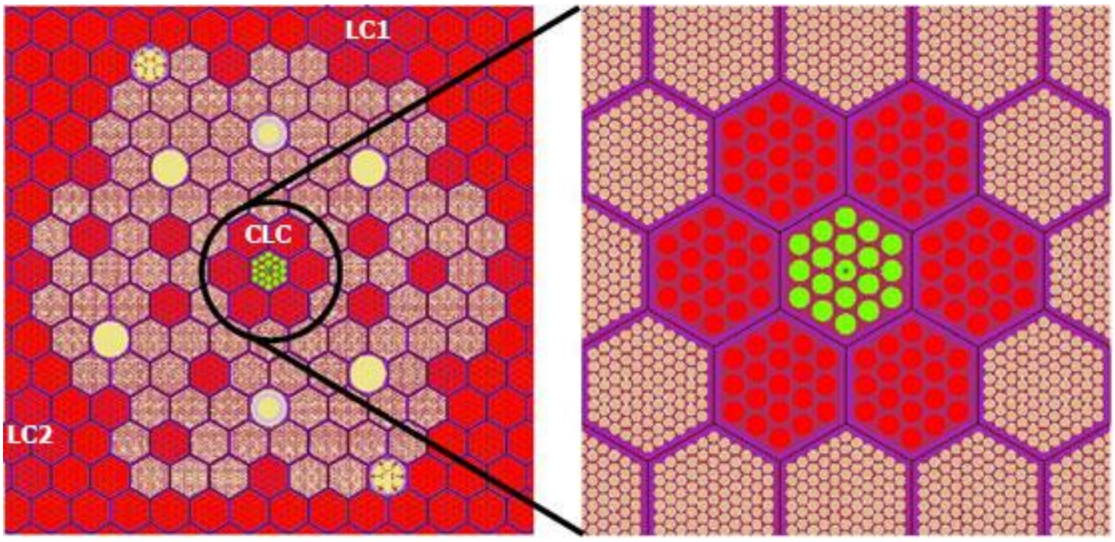

Fig. 4. Central loop layout of MBIR reactor

Taking into account total thermal power capacity (150 MWt), neutron flux axial distribution was estimated. Result are shown in Table III.

TABLE III: AXIAL DISTRIBUTION OF NEUTRON FLUX IN CENTRAL LOOP CHANNEL AT THE BEGINNING OF THE 3RD CYCLE.

Layer center axial coordinate, cm	MOX	MET	MET2a
25	3.21E+15	3.23E+15	3.23E+15
20	3.69E+15	3.70E+15	3.75E+15
15	4.04E+15	4.13E+15	4.15E+15
10	4.52E+15	4.53E+15	4.57E+15
5	4.73E+15	4.64E+15	4.79E+15
0	**4.98E+15**	**4.96E+15**	**4.86E+15**
-5	4.95E+15	4.85E+15	4.86E+15
-10	4.75E+15	4.66E+15	4.70E+15
-15	4.36E+15	4.26E+15	4.47E+15
-20	4.03E+15	3.79E+15	4.06E+15
-25	3.43E+15	3.22E+15	3.56E+15

In this way it is shown that MBIR core with 5.5 mm fuel pin can achieve the same neutron flux as initial variant with vibrio-packed MOX-fuel.

JARFR and ShIPR codes are used to simulate reactor life-time and to estimate reactivity drop. The last parameter is in good agreement with results obtained by means of ISTAR system taking into account all approximation for transport equation and Bateman equations (for burnup).

By means of JARFR and ShIPR codes fuel assembly power distributions were obtained for the core itself and radial shield. Axial distribution of linear heat load for the most loaded assembly was derived as a function of burnup. This data are used in thermo-hydraulic and stress-strain behavior analysis. Also these codes were used to estimate cross-sections required for radiation damage production rate calculation. These cross-sections were prepared in terms of 26 energy groups.

Taking into account characteristic neutron spectrum it is shown that radiation damage cross-section is 5% higher in “MET2a” option and 1% lower in “MET” option compared with initial “MOX” option. However all options demonstrates acceptable performance.

4. Simulation of MBIR Reactor Thermo-hydraulic Characteristics

This analysis is performed to estimate main thermo-hydraulic characteristics of MBIR-reactor. Several criteria were chosen to compare three options for MBIR reactor fuel. For MOX option the main criteria is cladding temperature, which should not exceed 610 – 620 degrees Celsius. For metal fuel options main criteria is pellet to cladding contact temperature, which should be below the level of eutectic formation. The highest pellet to cladding contact temperature is 575 degrees Celsius. In order to prevent significant thermal deformation of radial shield assemblies and fuel assemblies in in-vessel storage coolant heating in these elements should not exceed 80 and 40 degrees Celsius respectively.

Thermo-hydraulic reactor model was developed by means of Simulink [4]. Results obtained in terms of this model were compared with other thermo-hydraulic calculations based on independent channels model and with CFD-calculation carried out by means of ANSYS CFX [5]. By means of ANSYS CFX 3D reactor model was developed to provide cross-verification of Simulink model on the basis of NRC "Kurchatov Institute" data center [6]. Reactor itself is considered as a set of parallel channels with related throttlers. These channels are shown on FIG.5.

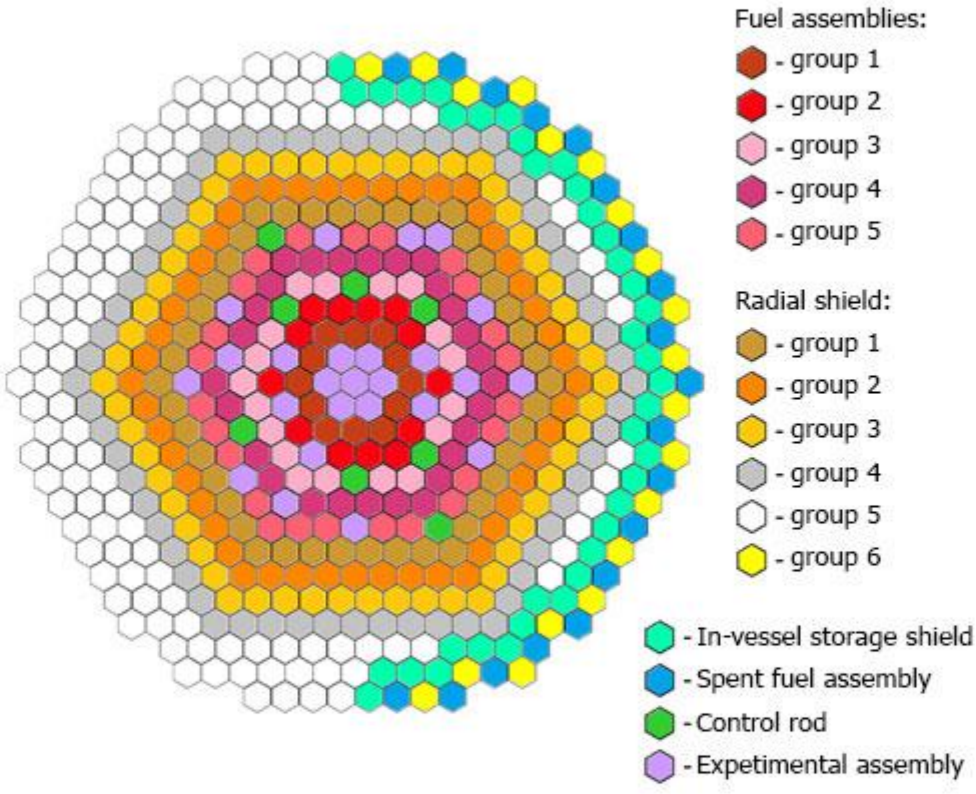

Fig. 5. Groups of channels

In case of 650 kg/s total flow rate "MOX" option with 6 mm fuel pin diameter demonstrates 420 kPa pressure drop between inlet and outlet of reactor. 13 % of total flow rate moves outside of fuel assemblies. Coolant heating in the core reaches 182 degrees Celsius. Average outlet coolant temperature is 512 degrees Celsius. Maximum temperature outside of the cladding is 541 degrees Celsius. Maximum temperature of pellet to cladding contact is 563 degrees Celsius.

In order to prevent significant deformation of radial shield assemblies and in-vessel storage fuel assemblies resulting from thermal stresses coolant heating can be decreased. In this case total flow rate was increased by 10% up to 715 kg/s. This extra 10% of flow rate are distributed among radial shield assemblies and in-vessel storage fuel assemblies to decrease coolant heating from 140 to 80 and from 80 to 40 degrees Celsius respectively. In this case coolant heating in the core decreased up to 164 degrees Celsius, reactor outlet temperature decreased up to 494 degrees Celsius. Pressure drop between inlet and outlet of reactor raised up to 445 kPa. 18 % of total flow rate moves outside of fuel assemblies. However, core flow rate stayed on the previous level, which means that maximum cladding temperatures did not change.

In case of 650 kg/s total flow rate "MET" option with 6 mm fuel pin diameter demonstrates 418 kPa pressure drop between inlet and outlet of reactor. 13 % of total flow rate moves outside of fuel assemblies. Coolant heating in the core reaches 182 degrees Celsius as in "MOX" option. Average outlet coolant temperature is 512 degrees Celsius. Maximum temperature outside of the cladding is 550 degrees Celsius. Maximum temperature of pellet to cladding contact is 572 degrees Celsius.

In order to prevent significant deformation of radial shield assemblies and in-vessel storage fuel assemblies resulting from thermal stresses coolant heating can be decreased for "MET" fuel option with 6 mm fuel pin diameter. In this case (as in previous one) total flow rate was increased by 10% up to 715 kg/s. As a result pressure drop between inlet and outlet of reactor raised up to 445 kPa. The same 18 % of total flow rate moves outside of fuel assemblies. Coolant heating in the core decreased up to 164 degrees Celsius, reactor outlet temperature decreased up to 494 degrees Celsius. Maximum temperatures outside and inside cladding stayed at the same level as in initial flow rate of 650 kg/s.

The last considered option is 5.5 mm fuel pin diameter core with total flow rate of 650 kg/s. As a result of decreased pin diameter pressure drop decreased up to 323 kPa. Only 12 % of total flow rate moves outside of fuel assemblies. Maximum temperature outside of the cladding is 530 degrees Celsius. Maximum temperature of pellet to cladding contact is 553 degrees Celsius. These temperatures are 15 – 20 degrees lower than for 6 mm fuel pin diameter options.

The opportunity to decrease thermal deformations of radial shield assemblies and in-vessel storage fuel assemblies was also investigated. For this purpose total flow rate was increased on 10% up to 715 kg/s. This flow rate increase results in drop of coolant heating in radial shield assemblies and in-vessel storage fuel assemblies below recommended level of 80 and 40 degrees Celsius. At the same time core coolant heating drops to 164 degrees Celsius. Reactor outlet temperature decreased up to 494 degrees Celsius. Maximum temperatures outside and inside cladding stayed at the same level as in initial flow rate of 650 kg/s.

For "MET2a" option with 5.5 mm fuel pin throttling refuse option was investigated. This option is considered in order to provide conditions for secure fuel assembly heat removal in case of possible excitations of energy generation density distribution resulting from fuel reloading and irradiation facilities heat generation. Main approach to provide secure heat removal is increasing of total core flow rate without any throttling resulting in uniform coolant distribution over fuel assemblies. However special condition should be met: pressure drop in this case should not exceed the same value in 6 mm fuel pin option with nominal flow rate. To meet this requirement total flow rate was increased to 40%. Radial shield assemblies and in-vessel storage fuel assemblies flow rate was increased to 10% to provide acceptable thermal stresses. As a result total core flow rate rose up to 940 kg/s and coolant heating dropped to 125 degrees Celsius. In order to keep the same level of fuel pin cladding temperatures core inlet coolant temperature was increased from 330 to 370 degrees Celsius. Average reactor outlet temperature reached 495 degrees Celsius. 16 % of total flow rate moves outside of fuel assemblies. Maximum temperature outside of the cladding is 546 degrees Celsius. Maximum temperature of pellet to cladding contact is 568 degrees Celsius.

Thus “MET2a” fuel option with 5.5 fuel pin diameter enables us to refuse from throttling in order to provide secure fuel assembly heat removal in case when maximum linear heat load does not exceed 48 kWt/m. This advantage is very useful for research reactor where energy generation density distribution can be strongly affected by fuel reloading strategy and local excitations of irradiated facilities.

All options considered demonstrate acceptable thermal conditions of fuel pin cladding. Outside fuel pin cladding temperatures for “MOX” options do not exceed 620 degrees Celsius. Pellet to cladding contact temperatures for metal fuel options do not exceed 575 degrees Celsius.

5. Comparative Analysis of Fuel Pin Stress-strain Behavior with MOX and Metal Fuel

All calculations to determine thermo-physical and thermo-mechanical behavior of fuel pin under normal operation conditions are carried out by means of TEGAS [7] code.

Simulation shows that vibrio-packed MOX fuel and U-Pu-10Zr fuel with sodium sub-layer options demonstrates approximately the same stress-strain behavior of fuel pin cladding because the main limiting factor here is ChS-68 steel swelling.

Stress-strain behavior analysis shows that metal fuel with sodium sub-layer and MOX-fuel options reach the same stress-strain conditions of fuel pin. Taking into account irradiation embrittlement of austenitic steel under high fluence and circumferential strain exceeding 2%, tangential stress in these two cases does not exceed maximum permissible value of 200 MPa. Performance of ChS-68 fuel pin cladding is experimentally confirmed by means BN-600 operation up to 80 – 87 DPA. Beyond 87 DPA fast degradation of strength properties takes place. Thus, these two options demonstrate acceptable performance up to 80 DPA.

According to [8, 9] there was no fuel pin failure with metal U-Pu-Zr fuel and sodium sub-layer up to burnup of 19% h.n. except cases with weld defects or increased cladding temperature.

Metal fuel with gaseous sub-layer option demonstrates cladding tangential stresses up to 400 – 500 MPa even on earlier irradiation stages, which is far beyond acceptable limit of 200 MPa for ChS-68 steel. Many experiments with high smeared density metal fuel also showed cladding failure.

6. Conclusion

The article presents results of studies of Multifunctional fast neutron research reactor MBIR core with metal fuel, which meets neutron flux requirements and has high level of functionality and research potential.

Two metal fuel options are considered and compared with reference option based on vibrio-packed MOX-fuel. It is shown that 6 mm diameter fuel pin with dense metal fuel and gaseous sub-layer can decrease reactivity drop over reactor life-time while keeping central loop neutron flux on acceptable level. 5.5 mm diameter fuel pin with metal fuel and sodium sub-layer provides the same neutronic characteristics (reactivity drop and maximum neutron flux in central loop channel) as in reference option with vibrio-packed MOX-fuel on the one hand and expands abilities for core formation specific for different experimental programs on the other hand.

For all three fuel options thermo-hydraulic parameters of the reactor core were estimated. All three fuel options with different total coolant flow rate demonstrate acceptable cladding temperatures. Maximum temperature outside of the MOX-fuel pin cladding does not exceed 620 degrees Celsius. Maximum temperature of the metal fuel pellet to cladding contact does not exceed 575 degrees Celsius.

Moreover, 5.5 mm diameter fuel pin with metal fuel and sodium sub-layer eliminates the need of throttling, which helps to provide secure heat removal for different core layouts under one restriction: linear heat load should not exceed 48 kWt/m. This advantage has especially high importance for research reactor, where energy generation density distribution can be strongly affected by fuel reloading strategies and local excitations of irradiated facilities. Therefore, MBIR reactor core with 5.5 mm diameter fuel pin with metal fuel and sodium sub-layer expands reactor potential as research facility due to improved core thermo-hydraulic.

Stress-strain behavior analysis shows that MOX-fuel and metal fuel with sodium sub-layer option demonstrates similar stress-strain behavior because of the same limiting factor of ChS-68 steel cladding swelling. Taking into account irradiation embrittlement of austenitic steel under high fluence and circumferential strain exceeding 2%, tangential stress in these two cases does not exceed maximum permissible value of 200 MPa. Thermo-mechanic analysis of high smeared density metal fuel with gaseous sublayer shows that tangential stresses can reach value of 400 – 500 MPa, which far beyond accepted limit.

As a result it was shown that on the basis of U-Pu-10Zr fuel pin with sodium sub-layer efficient MBIR core can be introduced to meet all requirements and expand consumer needs for this reactor as experimental facility.

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