Physical and technical basics of the concept of a competitive gas cooled fast reactor facility with the core based on coated fuel microparticles

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Abstract. The paper presents the results of development of the technical concept of a reactor facility with a high temperature fast helium cooled breeder reactor with expanded fuel breeding working in a closed fuel cycle and having its own role in the nuclear energy system due to efficient electricity generation, and, in the longer term, possibility of industrial applications. It is expected that the unit will have the level of specific capital costs comparable to competitive nuclear energy sources.

The concept of the reactor facility with the BGR-1000 reactor of 1000 MWe capacity is based on the synthesis of technologies of high temperature and light water reactors. Stage-by-stage development of the concept is assumed in terms of the use of BGR-1000 for electricity generation and industrial applications with a consistent increase of the core outlet coolant temperature.

The reactor design is based on the core with a fixed bed of coated fuel microparticles directly cooled by the helium flow. The core design allows to have limited excess reactivity, exclude significant radiological consequences of accidents, provide the required level of fuel breeding, as well as, in the longer term, the closure of the fuel cycle with respect for all actinides.

Key Words: fast helium cooled reactor concept, fuel microparticles, breeding, closed fuel cycle.

1. Introduction

The paper presents review of the concept of a reactor facility with the Gas Cooled Fast Reactor (GCFR) hereafter referred to as BGR-1000 [1-4]. The design of the 1000 MWe reactor facility is based on the core with fuel assemblies containing the fixed bed of coated fuel microparticles (CP) directly cooled by the helium flow.

Opportunities to use coated fuel microparticles were discussed in Russia in application for light-water reactor cores. Sufficiently high corrosion resistance of CP with silicon carbide (SiC) outer coating in water, steam and steam-water mixture is presented in several publications [5]. This fact initiated elaboration of a proposal to use flooding of a gas cooled reactor core with the cold water in accidental conditions as a protective measure.

It was also proposed to apply for GCFR a LWR metallic reactor vessel, e.g. similar to VVER-type reactors of Russian design, as well the loop-type arrangement of the primarily circuit. Such decisions allow to decrease the specific capital initial costs for reactor facility.

The need for elaboration of such a GCFR concept is also caused by probably demand of future nuclear power system for proposals on fast neutron reactors capable of operating in the closed fuel cycle [4].

2. General description

2.1.Basic requirements

Basic requirements to the reactor facility with the BGR-1000 fast reactor are the following:

- use of the known technical decisions avoiding significant expenses on R&D.
- exclusion of essential radiation consequences of accidents due to using of microfuel elements with multilayer protective coatings, corrosively resistant in the aqueous medium and preserving integrity at temperatures up to 1600^{o}C;
- features of neutron-physics: ensuring the core breeding ratio about 1.05 (or necessary reactor breeding ratio with optional use of axial and radial blankets), maximum reactivity margin for burnup about beta effective, small void reactivity effect, negative reactivity effect caused by flooding of the core by the cold water, possibility of achievement of the same characteristics in the closed cycle with the recycling of actinides.
- intention to decreasing initial capital costs of a power unit with BGR-1000;
- ensuring high thermal efficiency of the reactor facility due to decisions for steam turbines.

2.2.Reactor facility layout

The design of reactor facility with the fast helium cooled reactor 1000 MWe is based on the active core with fixed CP bed directly cooled by mainly radial (relatively to fuel assembly) flow of helium coolant up to 750^{o}C outlet temperature (at the initial development stage based on proven material properties).

An opportunity is considered to use fuel assemblies with various design decisions for providing of radial coolant flow profile through a layer of CP bed, which is distributed along the fuel assembly height to ensure necessary fuel performance and temperatures.

Different fuel types could be considered in the core concept. The basic design assumes the use of dense mixed uranium-plutonium carbide fuel composition. As a backup option, conventional dioxide could be considered; however, the decrease of core performance characteristics should be expected in this case because of relatively low fuel fraction in the core.

Basic conceptual solutions for the reactor facility with BGR-1000 are the following:

a) helium coolant with 16 MPa pressure and 750°C temperature at the reactor outlet;
b) the facility has a two-loop configuration and super-high steam parameters (30 MPa, 650°C);
c) the metallic reactor vessel;
d) block/loop-type arrangement of the basic equipment into a leak tight containment made of pre-stressed reinforced concrete;
e) the core consisting of the fuel assemblies (FAs) with fixed bed of fuel particles, cooled by cross-axial flow;
f) Control and Protection System (CPS) rods are moving in guiding tubes inside FAs;
g) steam generators (SG) have vertical direct-flow arrangement with steam generation in tubes (working fluid moves upwards);

h) refueling is performed with removing of the vessel cover, after the reactor was cooled and the core filled with water;

i) decay heat during refueling is removed by the Normal Flooding and Decay Heat Removal System (NFDHRS);

j) in events with fast loss of coolant, the core is cooled with water provided by Emergency Core Cooling System (ECCS), similar to one used in new-generation VVER designs, namely, by passive water supply from high- and low-pressure water tanks, and by active water supply from containment sumps (it is possible to launch the NFDHRS 24 hours after the accident begins);

k) in accidents with complete loss of power, decay heat is removed in the ambient air using the Passive Heat Removal System.

2.3.Principal technical parameters

The principal technical parameters of BGR-1000 are given in TABLE I.

TABLE I: BASIC TECHNICAL PARAMETERS OF BGR-1000 REACTOR INSTALLATION.

Parameter	Value
Reactor type	Fast helium-cooled
Energy conversion arrangement	Two loops
Electrical capacity, MW	1000
Thermal capacity, MW	2000
Primary circuit:	
Primary coolant	helium
Primary circuit pressure, MPa	16
Core inlet/outlet temperature, °C	350 / 750
Helium flow, kg/s	900
Loss of pressure, MPa, below	0.5
Secondary circuit:	
Working substance	Water-steam
Superheated steam pressure after SG, MPa	30
Superheated steam temperatureafter SG, °C	650
Feedwater temperature, °C	300
Steam pressure at intermediate reheater inlet, MPa	3.5
Steam temperature at intermediate reheater inlet/outlet, °C	310 / 650
Design lifetime, years	60
Average capacity factor per lifetime	0.87

Thermal energy is converted into electricity using a two-loop scheme. The primary coolant is high-purity helium, the secondary coolant is water-steam. In normal operating conditions, the main circulation gas blowers (MCGB) provide forced circulation of the primary coolant. Heat is transferred from the primary circuit to secondary one in vertical direct-flow steam generators. It is possible to increase the system efficiency by superheating the intermediate-pressure steam taken after the turbine high-pressure cylinder by the primary helium in the steam reheater – to 650°C (expected efficiency about 50 %).

The set of basic primary equipment includes: reactor (Fig. 1), three steam generators and one steam reheater, four main circulation pipelines of large diameter, and four MCGB. Steam generators and steam reheater are placed around the reactor perimeter, above the core level.

Electrically driven MCGBs are installed in the bottom part of each steam generator and steam reheater.

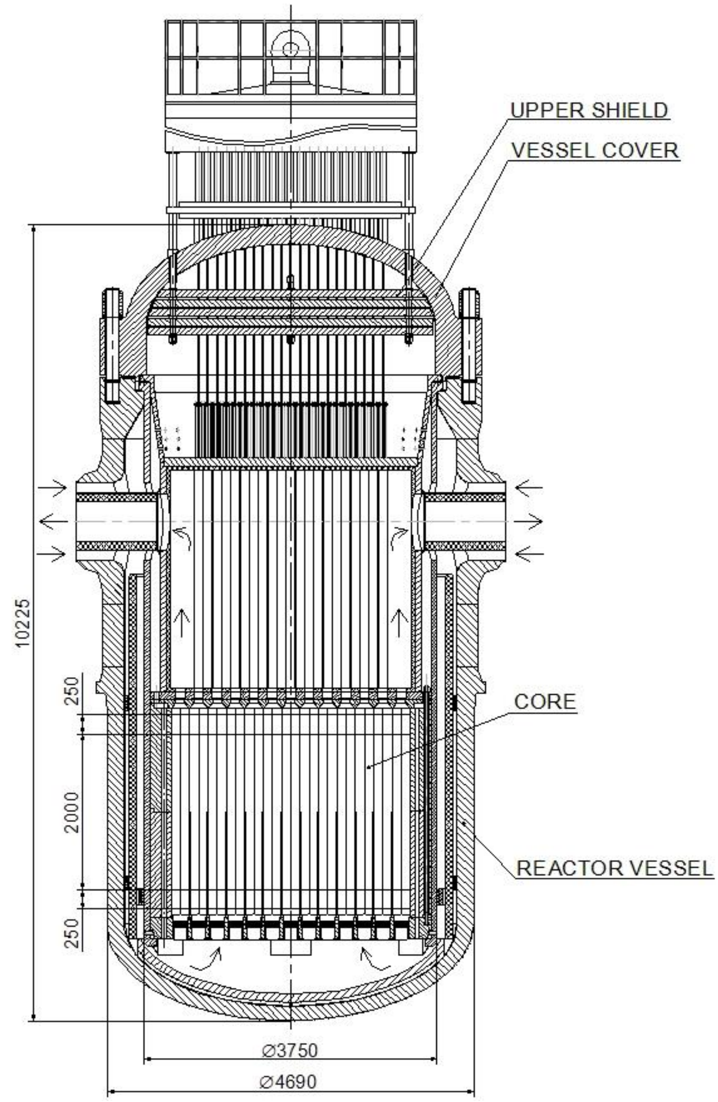

FIG. 1. Reactor (vertical cross-section).

2.4.Core

Based on the results of preliminary analysis of different options the following key provisions concerning the BGR-1000 core were adopted for further detailed development:

- Three-zone power profiling by plutonium content in the fuel;
- Hexagonal fuel assembly (FA) has the size across flats 150 mm;
- Top and bottom axial blankets are 250 mm thick each;
- Volume fraction of fuel is practically constant along the FA height;
- The core is surrounded by the radial blanket consisting of two rows of FAs;
- The reflected shield is installed after the second row of the radial blanket.

Spherical fuel microparticles with triple coating are used as fuel elements in the core and in the blankets. The outer diameter of the core CPs is assumed as 2 mm (CPs with close diameters has been already produced in Russia for research purposes). CP kernels are made of uranium and plutonium monocarbide with a diameter of about 1.64 mm. Selection of the carbide fuel on this stage of research was determined by its higher density compared with the oxide fuel, as well as by the fact that carbide fuel produces no free oxygen under irradiation, which would considerably increase gas pressure inside CPs at high burnup. These two factors are important for CP use in BGR-1000, because in a fast reactor the coating thickness is limited relative to those used in high-temperature graphite-moderated reactors. The first buffer layer of the coating is made of porous pyrocarbon (PyC) and has a thickness of about 0.125 mm. The second layer is made of dense PyC of about 0.005 mm thickness. The outer layer of the core CPs is made of silicon carbide (SiC) of about 0.05 mm thickness.

Core CPs are placed in between two coolant collectors. Core CP pebble bed height is 2.2 m, fuel pebble bed height in axial blankets is 2×0.25 m.

3. Performed studies

On the completed stage of the concept development basic qualitative and quantitative analysis of possible design decisions for the active core and first circuit equipment is performed. The reactor neutron-physics characteristics and their correlation with parameters of the core arrangement and fuel cycle options are preliminary evaluated. The possibility of core subcriticality in the flooded state was confirmed. Preliminary assessments are performed for thermal-hydraulic characteristics, analysis of the fuel behavior, etc.

3.1.Neutronic parameters

Neutronic calculations for BGR-1000 were performed with the complex of codes JARFR, which is based on the solution of the neutron transport equation in the multi-group diffusion approximation using a nodal scheme with the use of CONSYST/BNAB93 nuclear database.

At the initial stage of BGR-1000 conceptual development its operation was considered basing on the following assumptions. The reactor operates in a fuel cycle using reprocessed fuel of light-water reactors, which has low decay heat generation. "Fresh" fuel consists of depleted uranium and plutonium with the following isotopic composition:

Pu-238/Pu-239/Pu-240/Pu-241/Pu-242/Am-241 – 1.32/60.32/24.27/8.33/4.95/0.81 mass %.

This composition corresponds to plutonium produced in course of reprocessing of the enriched uranium fuel irradiated in a typical PWR of 900 MWe capacity to the burnup of 33 MW day/kg (the fuel is reprocessed after its 10-year cooling, and is placed in the reactor two years later). Depleted uranium with 0.1% mass content of U-235 is used for this fuel fabrication.

The BGR-1000 core, of 2200 mm height and 1360 mm equivalent diameter, has three power density profiling zones. Fuel assemblies of 150 mm turnkey size are positioned in a hexagonal lattice with a step of 151 mm. FA outer shroud has a form of an upturned perforated hexahedral prism made of stainless steel. The internal shroud, which is actually an outlet header, has a form of an upturned truncated cone made of Inconel alloy. For reactor reactivity regulation, each FA is equipped with three boron carbide rods. The tubes guiding rods, increase the strength of the FA structure. TABLE II presents the fuel particle parameters assumed for the optional core with three-zone profiling. TABLE III presents material fractions calculated for typical cells of the core, radial blanket, axial blankets, and other structural elements. In-core fuel assembly positioning scheme is shown in Fig. 2.

TABLE II: FUEL PARTICLE PARAMETERS ADOPTED FOR THE DESIGN OPTION WITH THREE-ZONE PROFILING.

	Fuel (U-Pu)C	**Buffer PyC-1**	**Dense PyC-2**	**SiC**
Thickness/ diameter, mm	0.820	0.126	0.005	0.05
Density, g/cm^3	12.0	0.8	1.8	3.2

TABLE III: FRACTIONS OF IN-CORE MATERIALS FOR THE DESIGN OPTION WITH THREE-ZONE PROFILING.

Material	**Core**	**Radial blanket**	**Bottom axial blanket**	**Top axial blanket**
Fuel (U,Pu)C	0.2455	0.3256	0.2455	0.2455
PyC-1 coating	0.1316	0.1746	0.1316	0.1316
PyC-2 coating	0.0060	0.0080	0.0060	0.0060
SiC coating	0.0637	0.0845	0.0637	0.0637
Steel	0.0223	0.0168	0.0223	0.0223
Inconel	0.0134	0	0.0134	0.0134
Helium	0.5175	0.3906	0.5175	0.5175

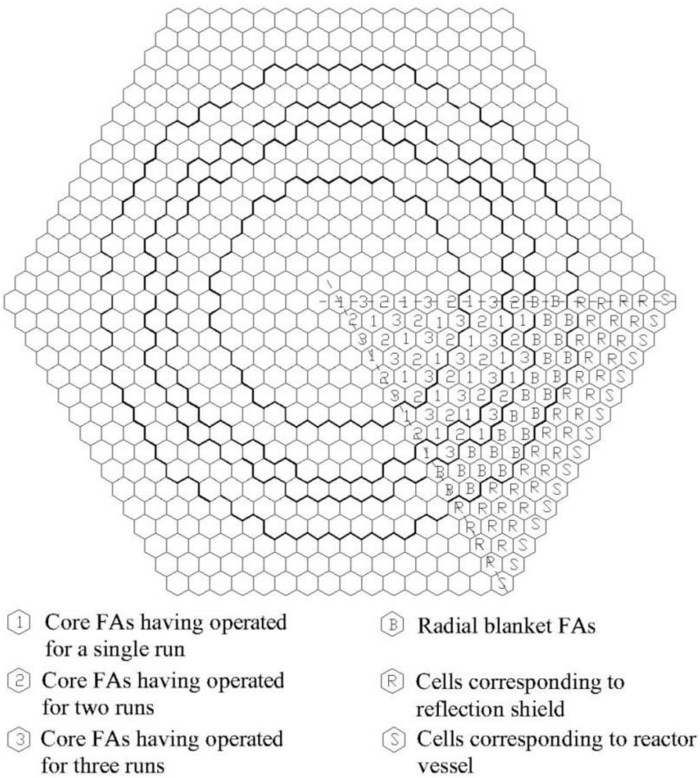

FIG. 2. In-core fuel assembly positioning scheme.

Neutronic calculations using the given compositions of fresh and burnt FAs (after one or two reactor runs) located in the core in accordance with the selected refueling scheme, were performed in order to provide a detailed simulation of the core fuel burnup. BGR-1000 core parameters for the case of its three-zone profiling are given in TABLE IV. In the equilibrium reactor refueling mode, reactivity change between fuel reloadings is less than the fraction of delayed neutrons, which is equal to 0.0037. The average core breeding ratio per a single fuel reloading cycle makes 1.059 at the beginning of operation, and 1.013 – at the end of fuel cycle.

TABLE IV: PARAMETERS OF THE BGR-1000 CORE DESIGN OPTION WITH THREE-ZONE PROFILING.

Parameter	Value
Core volume, m^3	12.82
Core height, m	2.20
Core effective radius, m	1.36
FA number in the core (core-1 / core-2 / core-3)	295 (121/114/60)
FA number in the radial blanket (2 rows)	138
Axial blanket thickness, m	0.25
FA positioning pitch, mm	151.0
Effective density of carbide, (U, Pu)C, g/cm^3	12.0
Plutonium contents in core-1 / core-2 / core-3, mass.%	10.5/16.5/20.0
Total heavy metal load of the core, kg	35951
Fissile plutonium load, kg	3640
Share of fissile Pu isotopes in the load, %	10.1
Total thermal capacity, MW	2000
Core thermal capacity, MW	1890
Axial blankets thermal capacity, MW	39
Radial blanket thermal capacity, MW	71
Core fuel lifetime, effective days	1800
RB fuel lifetime, effective days	600
Interval between reloading, effective days	600
Average fuel burnup in the core, % heavy atoms	9.7
Maximum fuel burnup in the core, % of heavy atoms	13.6

3.2.Some Reactivity Effect Issues

Principal issues related to the operation of the water flooding system, as a system influencing the reactivity have been studied in details.

In this regard, the reactivity effects resulting from the core flooding with water have been assessed, using detailed FA models and codes allowing to take account of both the cell macro heterogeneity determined by FA structural elements (headers, CPS rod tubes) and the micro heterogeneity determined by the structure of multi-layer-coated CP pebble bed (Fig. 3).

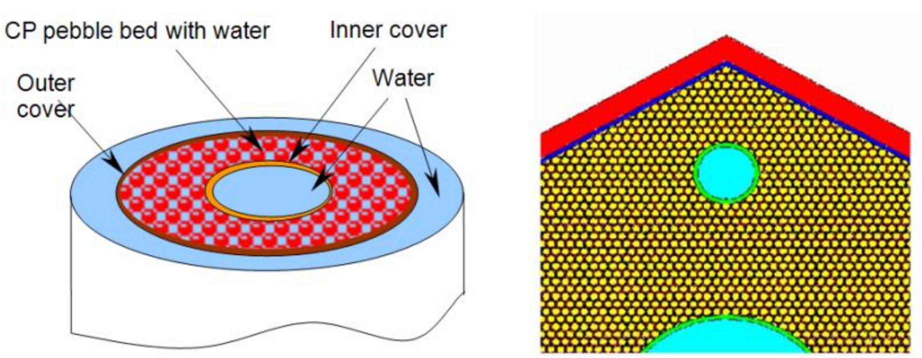

FIG. 3. Detailed fuel assembly models for studying water flooding effects.

Reactivity effects were estimated for the cases of core flooding with water steam and cold water, with considering the influence of the presence of water and steam in separate physical zones inside a FA: headers, CPS rod tubes, and CP pebble bed. The approach to assessing the emergency water flooding scenario was developed, and the efficiency of this measure as a means of reactivity influence was studied on the basis on calculated results. The calculated results presented here are obtained in conservative assumptions, i.e., for fresh fuel and without taking into account the fission products accumulated during irradiation. To guarantee the subcriticality (including the initial phase of flooding a fresh core with water), the solutions proposed earlier in the steam-water-cooled fast reactor design were considered, e.g. introduction in FAs of some permanently present efficient thermal neutron absorber, such as gadolinium. Assessments showed that this solution is also efficient for BGR-1000 fuel design, in case of minimum reactivity influence in the nominal mode of operation.

Fig. 4 shows the typical significant softening of the neutron spectrum in a FA in case of water flooding, which is the actual cause of the reactivity changes ensuring the possibility of efficient use of thermal neutron absorbers.

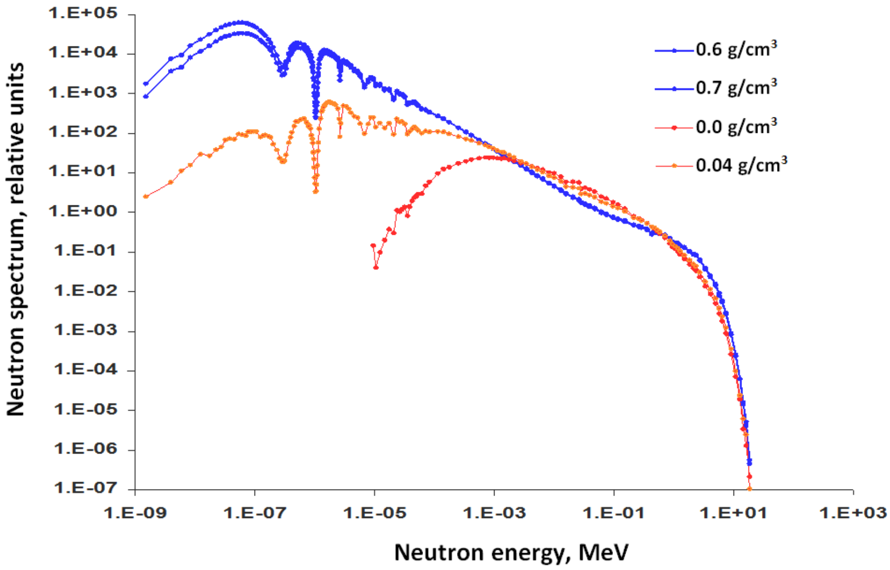

FIG. 4. Change of BGR-1000 fuel assembly neutron spectrum in case of water flooding.

3.3.Core thermal hydraulics

The accepted for design volumetric power rate (about 150 MW/m^3), thermal-physics properties of the helium coolant (rather low thermal conductivity and low density) together with specific design of fuel assemblies require taking a big care to the problem of modeling fluid dynamics and heat transfer in FAs and accurate estimation of uncertainty factors and overheating of structures and fuel.

For spatial simulation of the coolant fluid dynamics in the BGR-1000 core fuel assemblies it was used the Navier-Stockes treatment of the gas flow in FA collectors and porous body models in the fixed CP pebble bed.

The scheme of the flow in a fuel assembly and example of calculation results are presented in Fig. 5.

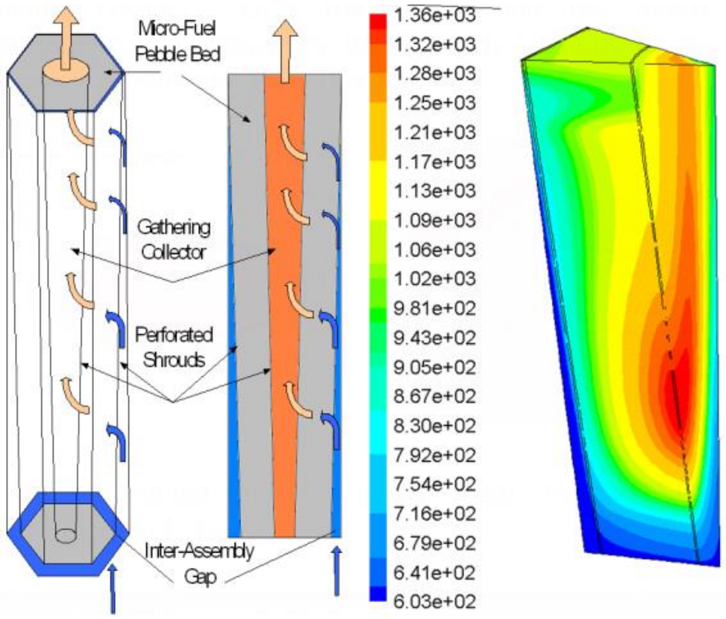

FIG. 5. Scheme of coolant flow in a fuel assembly, and typical temperature field, K.

In simulating of fluid dynamics in the BGR Fuel Assembly three 3D models and one combined (1D+2D) model have been tested and compared. The difference in the results obtained with the use of these models is in tolerable limits.

The features of flow in the fuel assembly are following:

- The values of coolant velocity in the FA pebble bed are by two orders of magnitude lower than in FA collectors;
- About 90-95% of total pressure drop in the fuel assembly is determined by drag forces in the inter-assembly gap;
- Maximum of the temperature is shifted relatively to the maximum of the power rate in the fuel assembly from 0.5 of its height down to 0.35;
- Sensitivity of temperature and other fluid dynamic fields in the FA to variation of geometry and friction in inter-assembly gap (which is feeding collector in the same time) is very essential;
- Temperature and pressure fields are much less sensitive to variation of friction in the CP pebble bed and, practically, not sensitive to variation of friction in the FA central collector.

3.4.Fuel particle design and performance assessment

The following factors were taken into account during the conceptual development of fuel particles for BGR-1000:

- GCFR using fuel particles should have the principal potential advantages of safety and possibility to assure the required fuel breeding ratio.
- Amount of energy obtained from a mass unit of fuel for the whole period of its stay in the core should be maximized (high burnup level per mass unit of fissile isotopes).

Fuel utilization should be acceptable, and fissile isotopes breeding in the core should ensure the reactivity margin, sufficient for compensation of reactivity loss due to burnup.

It was shown that, in the case of dense mono-carbide fuel and at maximum burnup level of about 14% the above mentioned CP dimensions ensure required neutronic parameters of the core and sufficient reactivity margin (with the average initial enrichment of ~10% of fissile isotopes).

Estimations showed that if usual dioxide fuel is used, the design reactivity margin should be increased.

4. Conclusions

The development of the reactor facility with gas cooled fast breeder reactor BGR-1000 possesses advantages from a synthesis of the proven technological decisions of high-temperature and light-water reactors.

In comparison with GCFR concepts with traditional container-type fuel elements (fuel rods), the concept of BGR-1000 reactor with fixed bed of coated fuel microparticles has the potential to eliminate essential radiation consequences of accidents due to inherent properties of fuel design.

The concept ensures the additional level of protection from the proliferation of the fissile materials. It is rather difficult to reprocess coated particles by the traditional aqueous methods, especially with the dense carbide fuel. In the same time, the reprocessing of coated particles is possible by the advanced high-temperature non-aqueous methods, for example, the method of volatile gas fluorides, developed in NRC «Kurchatov Institute».

The main directions of future investigations are the sensitivity analysis that should reveal possible issues requiring further qualification.

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