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Investigations for the substantiation of high-temperature nuclear power generation technology using fast sodium-cooled reactor for hydrogen production and other innovative applications (Part 1)

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Abstract

Neutronics and thermal physics studies of BN-VT reactor installation with 600-MW thermal power demonstrated the possibility in principle to achieve the required parameters of high-temperature fast reactor for production of large quantities of hydrogen on the basis, for instance, of one of thermal chemical cycles or high-temperature hydrolysis with high thermal efficiency of use of electric power. Relatively small dimensions, the type of coolant, selection of fissile material and structural materials allow developing nuclear reactor with particular inherent properties (exclusion of prompt-neutron reactor power excursions, removal of decay heat in passive mode) while ensuring enhanced nuclear and radiation safety.

Composition of BN-VT reactor facility includes sodium-cooled fast reactor, three cooling loops for emergency heat removal and three sets of equipment of the secondary cooling loop for heat transfer from the reactor to chemical installations generating hydrogen or to gas-turbine plant for supplying chemical equipment with electric power. Composition of each of the cooling loops includes intermediate heat exchanger arranged inside the reactor vessel, centrifugal pump and pipeline for removal and re-introduction of sodium in the reactor core. Contemporary requirements on the safety and financial performance of future generations of nuclear reactors were taken into consideration in the development of the reactor under study. Implemented calculation studies demonstrated that penetration of hydrogen within the limits of permissible allowances produce practically no effect on the neutronics and safety parameters of the reactor. Solution of the problem of fuel pin stability was mitigated due to the selection of low thermal load on fuel pins. Application of EP-912-VD steel as a possible optional structural material was examined.

Continued studies of heat-resistant materials and their behavior under irradiation are required.

Keywords: Fast reactor; High-temperature, Sodium; Hydrogen production; Pool-type configuration; Neutronics; Thermal physics; Safety issues; Steels.

Introduction

Nuclear power generation is not an alternative or a competitor within the general strategy of development of the fuel and power generation complex of the country, but, instead, it offers additional potential of preservation of efficiency of available fuel resources during extended periods of time and possibility to enhance reliability and safety of power supply becoming "the source of the source" of power and other re-

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sources. Several alternative strategies of development of nuclear power generation are currently under discussion [1,2]. One of the main requirements to future nuclear power generation technology – its large scale – implies enhanced level of safety of all its elements including reactor facilities and application of technologies of closed nuclear fuel cycle [3,4]. Development of innovative fast reactors with stressful temperature and dose rate loads with application of sodium as a coolant constitutes important direction of formation of the new technological platform [5,6].

The most significant problem determining future development of environmentally clean power generation is the inclusion of hydrogen in the fuel cycle. Hydrogen is a very attractive element as the replacement of oil and gas although by itself it is not a source but, instead, a carrier of energy. It is anticipated that the need in hydrogen production will be sharply increased in the nearest future. Currently the main method of hydrogen production is the methane reforming with steam. However, from the viewpoint of long-term perspective of large-scale hydrogen production, the above method is not viable since it requires consumption of non-renewable hydrocarbon resources and is accompanied with emission of greenhouse gases in the environment. That is why alternative methods of production of hydrogen with application of water splitting methods using thermal chemistry or electrolysis processes requiring high-temperature source of heat for enhancement of efficiency of the above processes are investigated [7,8].

Due to the application of such coolants as gases and liquid metals (sodium, lead) Generation IV nuclear reactors can serve as such sources of high-temperature heat [9,10]. Coolant temperature at the outlet of the core of such reactors can reach 900–950°C. This is a new class of nuclear reactors designed for both ensuring electric power generation with enhanced thermal efficiency (50%), and, as well, for the supporting the technological processes during hydrogen production, coal gasification and liquefaction, advanced oil cracking, conversion of biomass into liquid fuel, in chemical industry, metallurgy, etc.

Expenditure of energy is, without any doubt, needed for supporting such technological processes, but, as the result, the fuel obtained (using the example of hydrogen) possesses completely new quality allowing resolving numerous environmental problems.

Conceptual studies on the selection of the general outlook of high-temperature sodium-cooled fast power reactor (BN-VT) for large-scale nuclear and hydrogen power generation implemented at the AAC RF-IPPE under the supervision of V.M. Poplavsky demonstrated that [11] designing such nuclear reactor is a realistic task. Examination of existing reactors for the purpose of use of heat generated in them for the above described purposes is also attractive along with development of dedicated nuclear reactors for producing hydrogen. In that case heating of reactor coolant in separate loop to the required temperature must be implemented using electricity produced by such reactor. Such study was performed by the group of authors (Khorasanov et al.) on the basis of BN-600 reactor [12,13]. Addressing the issues of purely technological nature associated with high levels of temperature in the reactor facility (RF) becomes the first priority including the development of sodium coolant technology at elevated temperatures and high hydrogen concentrations during extended lifecycle, application of heat resistant and radiation resistant structural materials, ensuring corrosion resistance of such materials at oxygen concentrations present in sodium coolant at the level of 0.1 ppm becomes the first priority [14]. Discussion of complex (not only neutronics, but thermal hydraulics and technological) studies for substantiation of conceptual design and safety of 600-MW-th BN-VT high-temperature sodium-cooled reactor for production of hydrogen is the objective of the present paper.

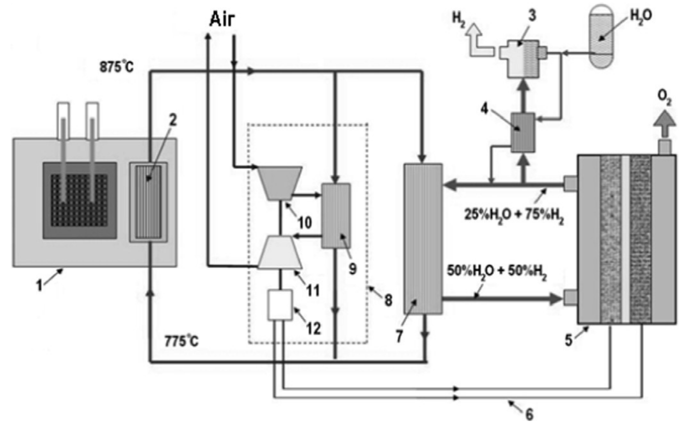

Fig. 1. Schematic layout of reactor facility for production of electricity and hydrogen on the basis of technology of solid oxide water electrolysis: 1 – fast reactor; 2 – intermediate heat exchanger; 3 – hydrogen separator; 4 – heat exchanger; 5 – solid oxide electrolysis cell; 6 – electrical power supply to the electrolysis cell; 7 – steam generator; 8 – gas turbine plant; 9 – heat exchanger; 10 – compressor; 11 – turbine; 12 – electrical generator.

Composition and techical characteristics of the BN-VT reactor facility

BN-VT reactor facility

Composition of BN-VT RF (Fig. 1) includes sodium-cooled fast reactor, three cooling loops of the emergency heat removal system, three sets of equipment of secondary cooling loops for transfer of high-potential heat from the reactor to hydrogen producing chemical installations or to gas-turbine plant intended for supplying electricity to chemical equipment. Each of the loops contains intermediate heat exchanger incorporated in the reactor vessel, centrifugal pump and pipeline for removal and re-introduction of sodium in the reactor.

Existing requirements on the safety and financial performance of reactors of new generations were taken into consideration in outlining the general configuration of the reactor under study. Innovative ideology of fast reactor on the basis of achievements and success of sodium-cooled fast reactor technology is further developed in it. BN-600 reactor successfully operated already during more than 30 years was chosen

Table 1
Main technical characteristics of BN-VT reactor.

Name of technical parameter, measuring unit	Parameter value
Rated thermal power, MW	600
Number of heat removal loops	3
Coolant temperature, °C	
– Core inlet	800
– Core outlet	900
– Intermediate heat exchanger inlet	775
– Intermediate heat exchanger inlet	875
Sodium flow rate through one intermediate heat exchanger, kg/s	1379
Absolute coolant pressure at the core inlet, MPa	≤ 1,0
Excess pressure within reactor gas volume, MPa	0,054

as the basis for the design of the reactor under investigation. Main technical characteristics of BN-VT RF are presented in Table 1.

Pool type configuration of main equipment of the primary (radioactive) cooling loop inside the reactor vessel consisting of the pressure vessel and secondary containment, which facilitates ensuring high level of safety and allows removing the boxes for arrangement of auxiliary systems of the first cooling loop. Pressure vessel of the reactor intended for installation of in-reactor equipment, sodium and argon of the primary cooling loop and arrangement of sodium circulation represents vertical cylindrical tank equipped with cone-shaped reactor vessel lid and elliptical bottom with support plate. From the inner side the reactor vessel is in contact with sodium with exception of its upper part (lid) working in contact with argon gas blanket of the reactor, and from the outer side with argon contained within the safety gap between the internal pressure vessel and outer safety containment.

Reactor core, intermediate heat exchangers, main circulation pump of the first cooling loop, emergency cooling heat exchangers, electrochemical hydrogen sensor, electrochemical oxygen and carbon sensor, pipeline for coolant supply core, gas compensation and overflow pipes and fuel cladding leak detection system are arranged inside the vessel of such nuclear reactor adapted for generation of heat. Because of their large dimensions cold traps (CT) are removed outside the reactor vessel.

Nuclear reactor characteristics

Based on the degree of preparedness it is intended to preserve as the initial step the existing reactor design and use uranium oxide fuel with changing only the temperature level. The main objective during this phase is to reveal the bottlenecks from the viewpoint of already well developed design leaving the issue of selection of structural materials open. Existing high culture of BN reactor design and technical solutions tested during many years of reactor operation should facilitate practical implementation of the reactor facility. During subsequent phases it is possible depending on the obtained results to address the possibility of use of other fuel compositions: MOX–fuel, mixed uranium–plutonium nitride fuel inside container type fuel rod, thorium fuel cycle and other potentially promising solutions requiring practical substantiation.

Table 2
Main characteristics of BN-VT reactor power unit.

Characteristic	Value
Power (thermal), MW	600
Nuclear fuel	UO_2
Core dimensions ($D\times H$) according to reactor vessel, mm	3900 × 1300
Reflector thickness, mm	200
Flat-to-flat dimension and wall thickness of hexagonal FA jacket, mm	96 × 2
Number of fuel rods in the FA	127
Material of FA jackets, fuel cladding and spacer wiring	EP-912-VD
Fuel cladding diameter and wall thickness ($d\times\delta$), mm	6.9×0.4
Cross-section dimension of spacer wiring, mm	
– For 91 central fuel rods	Ø1.05
– For 36 peripheral fuel rods	0.6 × 1.3
Fuel pellet sizes (bush), mm	
– Outer diameter	Ø5.9
– Internal diameter	Ø1.7
Reactor core height, mm	1030
Heights of axial blankets, mm	
– Top	300
– Bottom	350
Gas void height, mm	617
Total FA length, mm	3500
Time between reloading operations, days	330
Reload temperature, °C	230
Maximum fuel cladding temperature, °C	1025
Total temperature reactivity effect (230°C$\rightarrow T_{in}$) (230°C$\rightarrow$368°C)/(230°C$\rightarrow$800°C), %$\Delta K/K$	–1.431
Total reactor power reactivity effect ($T_{in}\rightarrow N_{rat.}$), %$\Delta K/K$	–0.452

Design of fuel assemblies, reactor core configuration and fuel loading map are similar for the BN-VT reactor to those for BN-600 reactor [15]. BN-VT reactor core consists of a set of fuel assemblies (FAs), CPS rods, neutron sources, boron control and steel control systems arranged in the reactor core according to triangular lattice with average spacing equal to 98.35 mm. Reactor core consists of 369 uranium loaded FAs with three types of enrichment, 27 CPS rods and two neutron sources. Along the radius the core is divided into three zones with different fuel enrichment. FAs contain sections of the top and bottom axial blankets consisting of pellets of depleted or natural uranium dioxide arranged inside the cladding common with fuel pellets. Assemblies of the radial blanket are arranged surrounding the reactor core.

It can be expected based on the correlation between power capacities of BN-600 reactor and BN-VT reactor under design that with reduction of thermal power from 1470 to 600 MW (by ~2.5 times) time interval between reactor core reloading can be increased from 140 days to one year – 330 days. Available efficiency of the reactor core shim system is expected to be sufficient with adequate safety margin for burn-up compensation, while increased temperature reactivity effect (isothermal reactor core heating from the reloading temperature to the coolant inlet temperature for the rated reactor power) can be compensated. Remaining reactivity effects should not significantly change. Characteristics of the reactor power unit are presented in Table 2.

Table 3
Variation of reactor reactivity versus concentration of hydrogen in the coolant.

Hydrogen concentration in the coolant, pcm	Reactivity gain introduced by hydrogen present in sodium composition, $\%\Delta C/C$
0	0.000
50	0.0081
100	0.014
150	0.022
200	0.027
250	0.032

Table 4
Chemical composition of high-nickel stainless steel EP-912-VD [20].

C	Si	Mn	S	P	W	Ni	Nb	Fe
0.03	0.32	0.06	0.005	0.005	9.13	35.97	0.93	Rem.

Some safety issues

Specific feature of reactor operation within the hydrogen production complex is the need to take into consideration the probability of hydrogen penetration inside the reactor core along the coolant circuit. The implemented calculation studies demonstrated (Table 3) that hydrogen penetration within the limits of permissible allowances produces practically no effects on the neutronics characteristics of the reactor and on safety parameters of the reactor.

Higher temperature level increases the probability of sodium boiling. Removal of sodium causes insignificant negative void reactivity effect explained by the use of uranium fuel. Thus, the requirement of significant increase of pressure in the primary cooling loop becomes not necessary. In order to organize closed fuel cycle there exist the possibility of examination of the use of uranium-thorium fuel cycle with close characteristics of reactivity effects. Key problem for high-temperature reactors is the fuel pin stability. The situation is somewhat mitigated for the proposed RF design due to the selection of low thermal load on the fuel pins. Maximum fuel burnup can be additionally reduced.

Structural materials

Selection of high-temperature material for particular reactor conditions is the most complex task from the viewpoint of reactor design. Alloys with high heat resistance and corrosion resistant in contact with sodium coolant at temperatures from 900 to 1200°C and radiation resistant up to 100 dpa are required as the material for fuel cladding. Results of studies of corrosion of structural materials are presented in Refs. [16–19]. Molybdenum and niobium alloys possessing attractive production properties and high temperature resistance along with corrosion resistance in contact with sodium coolant can be examined as such alloys.

Alloys of the basis of molybdenum can serve as the most suitable structural materials, but, however, their presence results in noticeable increase of neutron absorption which requires adjustment of fuel enrichment. According to preliminary estimations, increase of fuel enrichment in correspondence with maximum molybdenum concentration will not result, taking into account the available significant reactivity margin, in the violation of reactor safety requirements in the process of normal operation and during emergency situations. The problem of use of molybdenum-based structural material can be resolved by adjustment of fuel isotopic composition.

EP-912-VD steel was examined as possible optional structural material. This alloy with standard denomination X15H35B10Б (developed by the FSUE VIAM and the IPPE) is one of the promising structural materials for applications to work in contact with sodium coolant in air atmosphere at temperatures of 900–950°C. High short-term and long-term strength of the alloy is combined with high plasticity and viscosity characteristics at temperatures up to 950°C and at hot deformation temperature, stability of structure and of mechanical properties, good corrosion resistance in sodium coolant, as well as with high oxidation resistance at high temperatures. Argon electric-arc welding of sheets with thicknesses up to 12 mm is recommended to be performed using welding wire of XH60BT, 06X15H60M15 and X15И35B12 brands ensuring high resistance of metal welds against formation of hot cracks. Absence of molybdenum in the composition of steel is the important characteristic (Table 4).

Heat resistant 07X15H30B5M2 (CHS81) chromium–nickel austenitic steel developed by the «Prometey» Central Research Institute of Structural Materials (Table 5) is the alternative structural material. This steel is recommended for operation at temperature equal to 900–950°C. Studies of strength characteristics, corrosion resistance in sodium coolant and thermal stability performed at the «Prometey» Central Research Institute of Structural Materials demonstrated that the steel in question possesses the complex of physical, mechanical and technological properties required for its use in high-temperature nuclear reactors.

Comparison of reactivity gain introduced in the reactor by structural materials made of the above steels is shown in Table 6. Structural materials of BN-600 reactor core (cold-worked CHS-68 steel) introduce in the reactor core reactivity equal to–2,218•10^{-2} ($\Delta C/C$). This difference can be

Table 5
Composition of CHS81 stainless steel [21].

C	Si	Mn	S	P	W	Cr	Ni
≤ 0.07	≤ 0.2	0.8–1.2	≤ 0.01	≤ 0.015	4,5–5,5	14,0–17,0	29,0–31,0
Mo	Ti	Al	Other	Standard			
1.8–2.2	≤ 0.06	≤ 0.12	Cu ≤ 0.08; N ≤ 0.03; Fe ≤ ост; Y ≤ 0.05	TU14-1-3970-85 TU14-1-4244-87			

Table 6
Contribution of chemical elements in the composition of reactor core structural materials in the effective neutron multiplication factor, $C_{ef.}$

Chemical element	EP-912-VD		CHS-81	
	$\Delta K/K$	Nuclide composition (%)	$\Delta K/K$	Nuclide composition (%)
Fe	$-1.08\cdot10^{-2}$	25.9	$-8.78\cdot10^{-3}$	22.6
Cr			$-3.35\cdot10^{-3}$	8.6
Ni	$-1.67\cdot10^{-2}$	39.9	$-1.40\cdot10^{-2}$	36.0
Mo			$-3.12\cdot10^{-3}$	8.0
W	$-1.43\cdot10^{-2}$	34.3	$-8.42\cdot10^{-3}$	21.7
Mn			$-1.16\cdot10^{-3}$	3.0

compensated for by using the reactor CPS. Therefore, preference can be given to CHS-81 steel although the final choice can be made after implementation of comprehensive studies of different structural materials as applicable to high-temperature nuclear reactor.

Conclusion

Results of the implemented neutronics and thermal physics studies of 600-MW(th) BN-VT reactor facility demonstrated the possibility to achieve the required parameters of the high-temperature fast reactor for production of large quantities of hydrogen based, for instance, on the use of one of thermal chemical cycles or high-temperature electrolysis with high thermal efficiency and high efficiency of electric power generation (relative fractions of heat spent to cover auxiliary power consumption and for electric power generation can be determined subsequently on the basis of financial calculations) satisfying at the same time safety requirements. Relatively small dimensions, the type of coolant, selection of fissile material and structural materials allow developing the reactor with particular inherent properties (exclusion of prompt-neutron reactor power excursions, removal of residual heat in passive mode) ensuring enhanced nuclear and radiation safety and satisfying the requirements for Generation IV nuclear reactors

References

[1] A.A. Goverdovsky, S.G. Kalyakin, V.I. Rachkov, Teploenergetika 5 (2014) 3–9 (in Russian).
[2] V.I. Rachkov, S.G. Kalyakin, Izv. vuzov. Yad. Energ. 1 (2014) 5–16 (in Russian).
[3] V.I. Rachkov, Energosberezhenie i vodopod. 5 (2013) 2–8 (in Russian).
[4] V.I. Rachkov, Izv. vuzov. Yad. Energ. 3 (2013) 5–14 (in Russian).
[5] V.I. Rachkov, M.N. Arnoldov, A.D. Efanov, S.G. Kalyakin, F.A. Kozlov, N.I. Loginov, I. Yu Orlov, A.P. Sorokin, Teploenergetika 5 (2014) 20–30 (in Russian).
[6] V.I. Rachkov, S.G. Kalyakin, O.F. Kuharchuk, I. Yu Orlov, A.P. Sorokin, Teploenergetika 5 (2014) 11–19 (in Russian).
[7] International Atomic Energy Agency, Hydrogen as an Energy Carrier and its Production by Nuclear Power, IAEA-TECDOC-1085, IAEA, Vienna, 1999.
[8] A.V. Morozov, A.P. Sorokin, in: Proceedings of the 21th International Conference on Structural Mechanics in Reactor Technology (SMIRT-21), a Seminar on High-Temperature Projects, Kalpakkam, India, 2011 14–15 November (in Russian).
[9] NEA, Innovation in Nuclear Energy Technology, OECD Nuclear Energy Agency, 2007 No. 6103.
[10] E.S. Albitskaya, At. Teh. Rubezhom 11 (2013) 3–16 (in Russian).
[11] V.M. Poplavsky, A.N. Zabudko, E.E. Petrov, At. Energiya 106 (3) (2009) 129–134 (in Russian).
[12] G.L. Khorasanov, V.V. Kolesov, V.V. Korobeynikov, Izv. vuzov. Yad. Energ. 2 (2015) 81–87 (in Russian).
[13] G.L. Khorasanov, A.P. Ivanov, A.I. Blokhin, Altern. Energ. 6 (2004) 57 (in Russian).
[14] S.G. Kalyakin, F.A. Kozlov, A.P. Sorokin, in: Proceedings of the 21st International Conference on Structural Mechanics in Reactor Technology (SMIRT-21), a Seminar on High-Temperature Projects, Kalpakkam, India, 2011 14–15 November (in Russian).
[15] Yu.A. Kazansky, M.F. Troyanov, V.I. Matveev, At. Energiya 55 (1) (1983) 9–14 (in Russian).
[16] B.A. Nevzorov, V.V. Zotov, V.A. Ivanov, O.V. Starkov, N.D. Kraev, E.B. Umnyashkin, V.A. Solovyov, Korroziya Konstrukcionnykh Materialov v Zhidkikh Schelochnykh Metallakh [Corrosion of Structural Materials in Liquid Alkali Metals], Atomizdat Publ., Moscow, 1977. (in Russian).
[17] N.M. Beskorovajny, A.G. Yoltuhovsky, Konstruktsionnye Materialy i Zhidkometallicheskie Teplonositeli [Structural Materials and Liquid Metal Coolants], Energoatomizdat Publ., Moscow, 1983. (in Russian).
[18] J. Zhang, T.F. Marcille, R. Kapernick, Corrosion 64 (7) (2008) 563–573.
[19] A.W. Thorley, in: Proceedings of the 4th International Conference on Liquid Metal Engineering and Technology, Avignon, France, 1988.
[20] A.G. Koltsov, V.V. Roschupkin, M.M. Lyahovitsky, N.L. Sobol, M.A. Pokrasin, Eksperimentalnoe Issledovanie Fiziko-Mehanicheskikh Svojstv Konstruktsionnoj Stali EP-912 [Experimental Research of Physics and Mechanical Properties of Structural Steel EP-912]. Journal of East Ukrainian National University named after Vladimir Dal, Moscow. Available at: http://archive.nbuv.gov.ua/portal/soc_gum/vsunu/2011_12_1/ Kolcov.pdf: (in Russian).
[21] Metally i splavy, marki i khimicheskiy sostav, Sostavitel i redactor, I.V. Bekkerev, Metals and Alloys: Brands and a Chemical Composition, in: I.V. Bekkerev, Ul yanovsk (Eds.), UlSTU Publ., 2007 (Eds.). ISBN 978-59795-0042-3. Available at: http://www.bibliotekar.ru/spravochnik-73/index.htm: (in Russian).