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Physical characteristics of large fast-neutron sodium-cooled reactors with advanced nitride and metallic fuels

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Abstract

Mixed nitride uranium–plutonium fuel is the most attractive and advanced type of fuel for fast-neutron sodium-cooled reactors, and is considered as the base fuel for future commercial fast-neutron power reactors. However, a substantial increase in breeding achieved thanks to the use of this fuel instead of oxide fuel is not enough to meet new requirements the major of which, minimization of the burn-up reactivity margin, is defined by the reactor core breeding. This paper presents the results of computational studies aimed at choosing the best possible layout for the core of a large fast-neutron sodium-cooled reactor meeting modern requirements.

Metallic fuel has been considered as the fuel for fast-neutron reactors since their early designs due to high density and heat conductivity and the smallest possible number of the diluent nuclei which provides for the highest possible breeding efficiency. The paper presents the features of the fuel under consideration and the results of computational studies into its application in large fast-neutron sodium-cooled reactors as compared to nitride fuel. A conclusion is made that, with the same design of the reactor core, metallic fuel is inferior to nitride fuel from the point of view of ensuring the safety of the reactor facility.

Most of the calculations were performed in a diffusive approximation based on the TRIGEX software package.

Keywords: Fast sodium reactor; Nitride uranium–plutonium fuel; Metallic fuel; Fuel volume fraction; Burn-up reactivity margin.

Introduction

At the present time, mixed uranium–plutonium nitride fuel is considered as the base type of fuel for future fast-neutron commercial power reactors, specifically for the BREST-300 and BN-1200 reactors. Along with a high breeding ratio, this fuel features increased density and heat conductivity, as well as good compatibility with liquid-metal coolants and cladding materials, especially in emergencies.

In Russia, uranium nitride (UN) fuel was used only in the core loads for the BR-10 experimental reactor [1]. Since 1970, the BR-10 reactor has been used for irradiation of experimental nitride-fuel assemblies manufactured using different technologies, with different porosity and two types of contact underlayers (sodium and helium). These studies formed the basis for the development of two full reactor core loads with uranium mononitride fuel assemblies (~200 FAs), in which a burnup of up to 8.7% h.a. was achieved. Later on, in 2000–2005, fuel elements with uranium-plutonium nitride fuel were irradiated in the BOR-60 reactor as part of the BORA-BORA experiment conducted in collaboration with CEA, France, [2]. The maximum burnup achieved in the BORA-BORA experiment was 12.1% h.a.

Metallic fuel has been considered as the fuel for fast reactors since their early designs due to increased density and thermal conductivity, and the largest possible number of the diluent nuclei which provides for the highest possible breeding ratio. This fuel was used in the first US sodium-cooled fast-neutron reactor, Fermi [3]. A similar design, BN-50, was developed in our country in 1965 but has never been implemented.

In the USA, this fuel was considered not from the point of view of breeding, but due to cheap technologies of the

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fuel fabrication (casting) and reprocessing (electrochemistry) in a closed fuel cycle, along with its high (inherent) safety. An economic analysis has actually shown that a reactor fuel cycle with such fuel (as compared to ceramic fuel, its powder fabrication technology and water radiochemistry) turns out to be seven times as cheap.

One drawback of metallic fuel is intensive interaction of fuel elements with the steel cladding in high-temperature conditions. At a temperature of ∼560 °C, plutonium reacts with the steel components (iron, nickel, chromium) to form a eutectic liquid compound, which, if formed at the inner cladding boundary, is capable to cause the fuel cladding to break down just in several hours. Addition of zirconium (∼10% wt) to metallic fuel increases the eutectic formation temperature by about 80 °C, thus making the fuel elements serviceable under the temperature conditions acceptable for the fuel column. For this reason, a ternary alloy, U–Pu–Zr, was proposed for use in fast-neutron reactors. However, the overall temperature level in fast-neutron reactors with metallic fuel is still lower than in reactors with ceramic fuel (by approximately 60–80 °C), resulting in a lower thermodynamic efficiency. All this was found out during studies at the EBR-II reactor in the USA. Metallic fuel has a fairly low melting point, and is so effective only when used with a sodium contact underlayer.

A positive feature of metallic fuels using electrochemistry is that in the deposited at the cathode the uranium and plutonium content of the zirconium is stable and is ∼10%.

A helium contact underlayer leads to a temperature growth to beyond the melting point and requires the heat density to be reduced considerably. Historically, uranium molybdenum alloys (U+7% Mo and U+10% Mo) were used in early fast-neutron reactor designs. They were employed in the experimental reactors DFR, Great Britain, and Enrico Fermi, the USA. The EBR-II reactor used metallic fuel based on a ternary alloy (U-Pu-Zr). In 1980s in the USA, the PRISM fast reactor with metallic fuel based on a ternary alloy with a relatively low electric power (400 MW) was designed.

A more detailed comparison of uranium–plutonium mixed nitride fuel and metallic fuel from the point of view of "inherent safety" is presented in [4].

Nitride fuel

Nitride fuel for fast power reactors is the next prospective fuel after oxide fuel. As compared to oxide fuel, nitride fuel provides for a higher breeding ratio, specifically in the reactor core, while its high thermal conductivity ensures a higher safety of the facility thanks to a greater temperature margin to melting [5]. Thermal conductivity of nitride fuel is about seven times as high as that of oxide fuel [6]. Where required, this makes it possible to increase the linear power up to ∼70 W/cm. Nitride fuel is a rigid fuel and has internal porosity (∼15%) which is formed in the process of fabrication. The service life of such fuel depends on its swelling under the reactor conditions, resulting in a loss of the cladding integrity when the clearance between the fuel cladding and the fuel column is taken up in the course of irradiation.

Table 1
Comparative characteristics of nitride-fuel core options with breeding or steel screens.

Option	Breeding screens	Steel screens
Side screen and bottom screen	UN	Steel
Maximum fuel burn-up, % h.a.	11.2	11.4
Reactivity change, % $\Delta k/k$	−0.43	−0.68
Sodium void reactivity effect, %$\Delta k/k$	0.20	0.35
Maximum linear power, kW/m	47.1	47.9
Maximum FA power as begin of cycle/end of cycle, MW	8.7/8.75	8.54/8.56

(В таблице 1 и ниже приводятся смысловые названия характеристик – закрашено жёлтым, что является полезным для публикации)

There are no reasonably accurate and credible methods to calculate the nitride fuel swelling as a function of various parameters, including the fabrication technology. However, such procedures and programs are developed by researchers based on the existing experimental data by modeling major processes involved in the nitride fuel swelling during irradiation.

Nitride-fuel reactor core. Replacement of breeding screens for steel screens. TRIGEX, a software package, was used for neutronic calculations [7,8]. All calculations were performed for the steady-state conditions of uniform off-line refueling characterized by an equal number of the core fuel assemblies being replaced in a single refueling process and by equal refueling intervals.

A model of a large fast-neutron sodium-cooled reactor with nitride fuel was described in detail in [9–11]. The considered reactor core consisted of 432 fuel assemblies, each of which contained 271 fuel elements with a diameter of 9.3 × 0.6 mm. The reactor core was surrounded by a bottom end screen (BES) and a side blanket region (SBR). A mixture of uranium and plutonium mononitrides (U–Pu)N with a density of 11.5 g/cm^3 was considered as the fuel for a large sodium-cooled fast-neutron reactor, and depleted uranium mononitride (UN) was considered as the breeding material.

The main idea behind the abandonment of breeding screens consists in avoiding the generation of low-background plutonium, that is, contributing to the proliferation resistance. Physical characteristics of a uranium plutonium nitride fuel core with breeding and steel (side and bottom end) screens are compared in Table 1.

As can be seen from the presented data, the abandonment of breeding screens leads to an increase in the fuel burnup reactivity margin and in the sodium void reactivity effect (SVRE). Fig. 1 shows the reactivity change for a 330-day period after the breeding screen replacement for steel screens.

Enlarged reactor core (468 FAs) with increased fuel fraction and breeding screens. An increase in the fuel volume fraction leads to a reduction in the burn-up reactivity margin with no decrease in the SVRE value. To avoid the need for changing the reactor's key design parameters relating to the FA flat-to-flat dimensions, an increase in the fuel element diameter should be accompanied by a reduction in the number of the fuel elements in the FAs by one row. To compensate