# Fuels for advanced sodium cooled fast reactors in Russia: state –of-art and prospects

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***International Conference on Fast Reactors and Related Fuel Cycles (FR09)***

***7-11 December 2009, Kyoto, Japan***

# Presentation Outline

- ✓ **Background: SFR technology and fuel**
- ✓ **Advanced SFR cores (BN-800, BN-K)**
- ✓ **Principal results of advanced fuels study**
- ✓ **Summary**

# Current Status of SFR Technology in Russia

**Russia (USSR): the best results of SFR technology assimilation**

**~ 140 reactor-years: ~ 35% of SFR-years world wide**

Experimental facilities

Advanced design

**BR-10** (1959)
**BOR-60** (1969)
→ **BN-350** (1973)
→ **BN-600** (1980)
**BN-800** (under construction)
→ **BN-1800** (conceptual design)
**BN-1200** (under development)

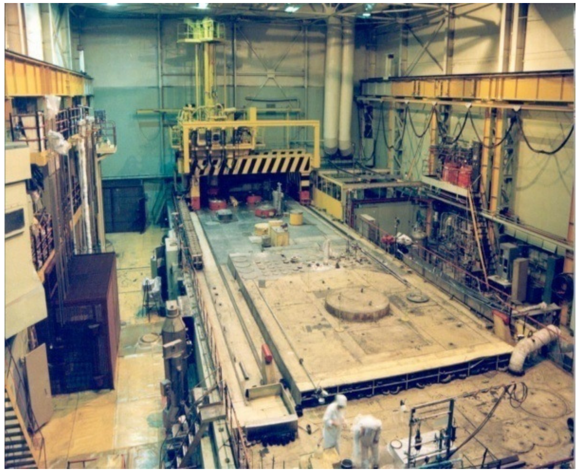
***Different fuels ($PuO_2$, UC, UN, UPuN, $UO_2$, $UPuO_2$, alloyed and non-alloyed metallic fuels, inert-matrices fuels) have been irradiated and investigated***

# BR-10, BOR-60 Fuels

BR-10 standard fuel
- PuO2
- UC
- UN

BOR-60 standard fuel :
- MOX vibro (since 1980)

Experimental fuel:
- Me alloyed, unalloyed
- Carbide UC, UPuC
- Nitride UN, UPuN
- Carbo-nitride UCN,
- Inert-matrices fuels,
- MOX pellet

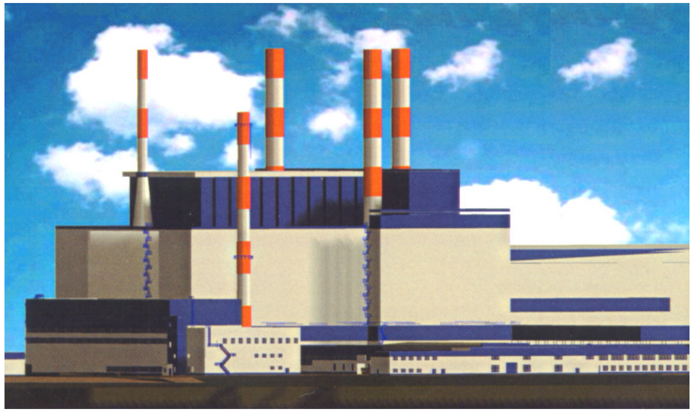


# BN-600 Fuels

- **Standard fuel**
  **$UO_2$ pellets**

- **Extensive testing of MOX fuel (pellets and vibro) and cladding materials**

# BN-800



## Standard fuel - MOX vibro

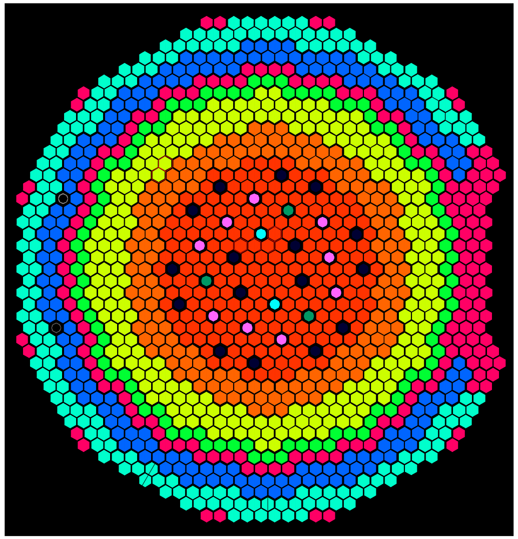
- **Wrapper steel:**

**EP-450 (13Cr-2Mo-Nb-P-B)**

- **Cladding steel:**

**ChS-68 cw (16Cr15NiMoNbMnTi)**

- **Max burn-up** 10at%
- **Max dose** 90 dpa



***In frame of National Program the works on ChS-68cw irradiation stability improving are carried out ;***

***1-st irradiation stage of experimental fuel pins with cladding made of EK-164cw (16Cr-19Ni) is successfully completed in BN-600. The aim is max dose of 110dpa for BN-800 core.***

# BN-K reactors : BN-1800, BN-1200

The basic BN-K core design requirements:

- Closed fuel cycle with minimum amount of radwaste
- Core breeding ~ 1 ---> decreasing of reactivity excess per fuel burn-up and compensation system efficiency up to (<βeff);
- Breeding ratio up to 1.45 for dense fuels;
- High burn-up and increased operation cycle for economy improving.

Reference fuel is MOX. Core breeding for MOX core is provided by higher fuel volume fraction and increased fuel smeared density up to 9,2 g/cm$^2$. But the better physics parameters are provided by a mixed nitride cores.

The nitride core is more compact, has higher core breeding and breeding ratio, permits to decrease the excess reactivity not only per burn-up, but per temperature-power effects compensation.

However additional studies are required in order to prove the possibility of nitride high burn-ups and nitride core safety. Nitride fuel is the farther option.

# BN-K reactors: structural materials

- **Wrapper - EP-450 (13Cr-2Mo-Nb-P-B)**

- **Cladding - ferritic -martensitic steels.**

There is some experience with EP-450 cladding in BOR-60, BN-350, BN-600. Experimental pins with EP-450 cladding and vipac MOX have been successfully tested in BOR-60 up to 142 dpa. Low level of EP-450 high temperature strength restricts its application for cladding in BN-600 and BN-800.

- **Complex-alloyed EK-181, ChS-191 steels** with higher level of high temperature strength are under investigation.

First of all, these steels are characterized by additional alloying of C, N, W, Ta and some decreasing of Cr content.

- **Their ODS-modifications** are under investigation aiming high temperature stability increase.

The mastering of fabrication process and complex out-of-pile investigations of ODS steels tubes of ferritic (on EP450 base ) and martensitic (on EK181 base) classes are under way. Preparations are in progress of steels samples irradiation in BN-600 material assemblies (MA) to maximum dose of ~ 160 dpa. ODS steels should provide the increase of burn-up value up to ≥20at% and temperature parameters.

# BN-K reactors : design development

## BN-1800

- **Reactor conceptual design** with pellet MOX and improved ferritic-martensitic steels with max burn-up

  17at% - I stage

  20at% - II stage.

  In order to use ferritic-martensitic steels the max cladding temperature is decreased to 660-670$^0$ C (instead of 700$^0$ C in BN-600, BN-800). It is achieved by means of sodium temperature rise decreasing in the core (up 140$^0$ C).

- **Technical proposal** for the reactor with nitride core (max burn-up – 13at%, max dose 160 dpa).

  The transition to mixed nitride may be done without changing of oxide core and fuel assembly designs. The smeared density of nitride with natural nitrogen is ~80%.

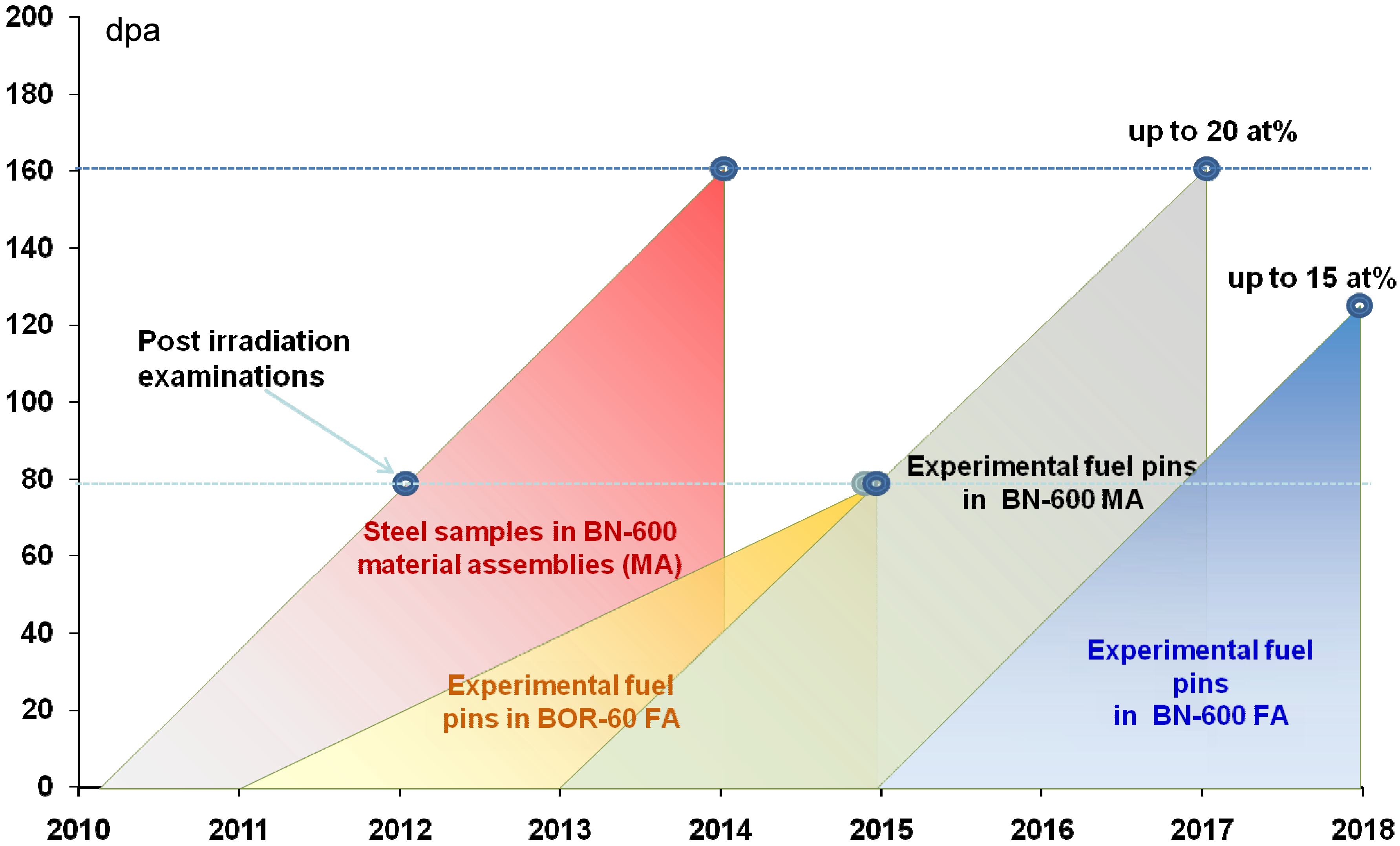
## BN-1200

- R&D work is under way with maximum using of already time-tested and scientifically based technical decisions, realized in BN-600 and BN-800 designs. **Core concept and burn-up level are that of BN-1800.**

- Possibility of MA utilization is assumed also. Two options of MA utilization are under consideration: homogeneous type with small MA addition in standard fuel and of heterogeneous type (in special targets).

- In order to increase breeding ratio, the heterogeneous cores are considered with depleted alloyed or unalloyed metals. One of the oxide-metal core models with axial heterogeneity fits well into the designed BN-1200 core without some changing and has high breeding ratio (~1.4) and almost zero changing of reactivity per burn-up.

# Reactor tests to validate BN-1200 fuel performance

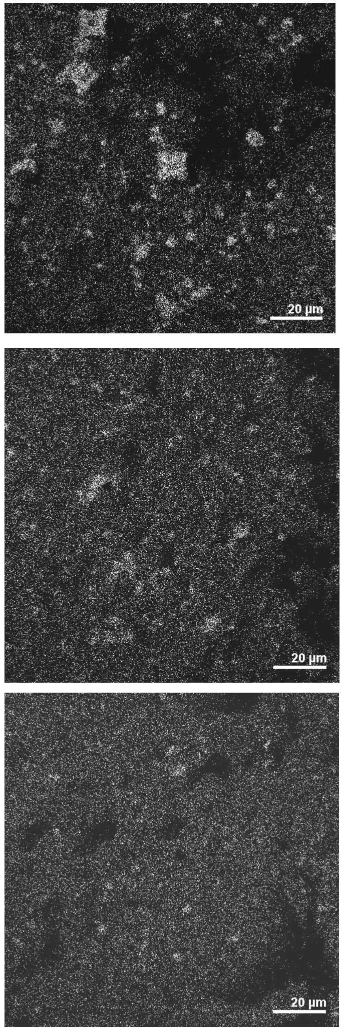


# Principal results of fuels study: MOX pellets

Pellet MOX fabrication methods have been developed in VNIINM, Moscow. Two methods for fuel powder preparation: mechanical mixing of initial $PuO_2$ and $UO_2$ (MMO method) and mixed oxide co-precipitation (GRANAT method).

The major problem of MMO methods is to achieve the blend homogeneity. Russian process of homogenous powder mixture fabrication is named as 'Vortex Milling Process' (VR-process). Pu distribution uniformity in fuel pellet is comparable to chemically co-precipitated fuel. Powder takes on special properties contributing to more active sintering of pellet compressed. VR-process is universal and permits to fabricate both oxides and nitrides of required quality.

*(Photos of the blend samples in characteristic X-radiation. The milling-blending time increases from up to bottom)*




The experimental fuel assemblies with pellet MOX fuel have been irradiated in BN-600 reactor. The maximum burn-up is ~12at%.

# Principal results of fuels study: MOX vibro

The fabrication technique of vibropack MOX fuel has been developed in RIAR, Dimitrovgrad. Vibropacking fabrication technique and pyrochemical reprocessing permit to realize the remote- automatic type of granulate and fuel pins fabrication. The required quality granulate is received by combined electro-crystallization of $UO_2$-$PuO_2$ from molten alkali metal chlorides. The technology of vibropack MOX fabrication by mechanical oxides mixing is developed also. The method enables to provide the predetermined Pu distribution and MA inclusion

The experimental fuel assemblies with vibro MOX have been irradiated in BN-600 reactor. The maximum burn-up is ~10.5 at%.

# Principal results of fuels study: Nitride

- **BR-10 reactor: 2 loadings with UN (660 and 590 fuel pins).**

**18 years operation of BR-10 with UN fuel has shown its good performance up to 8 at%. More than 99 % of fuel pins have reached the design value of (8 at %) w/o failures.**

**For 2-nd core (8.8 at%) number of fuel failures has increased (24 cases), mainly, at burn-up > 8at%. The reason is fuel-cladding mechanical interaction (FCMI).**

**PIEs of 6 FA : data on swelling and gas-release from nitrides in dependence on burn-up have been received.**

- **BOR-60 reactor:**

**1) tens pins with UN, max burn-up > 8 at%,**

**2) few pins with UPuN, max burn-up 4 at %, 8.95 at %.**

**3) 4 pins with UPuN with increased Pu content (45 % and 60 % ) within the frame of joint Russian-French BORA-BORA experiment. Maximum burn-up – 12 at%.**

**All pins are helium -bonded.**

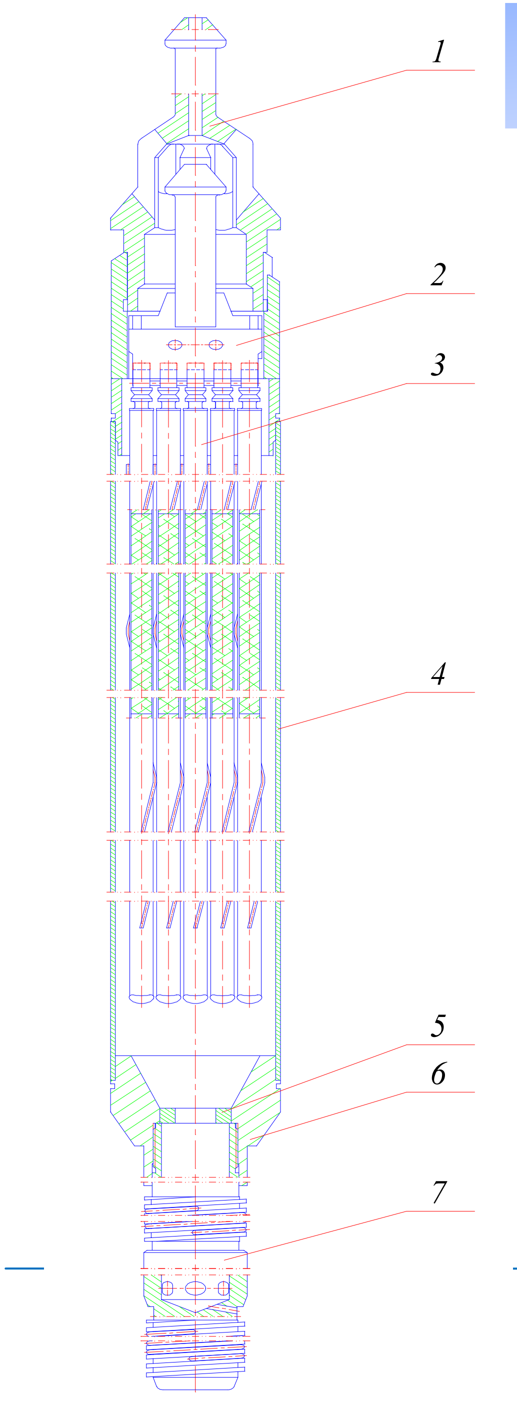
# BORA-BORA experiment: fabrication, irradiation in BOR-60 and PIE of high Pu content fuels

**Start of fabrication- 1996,**

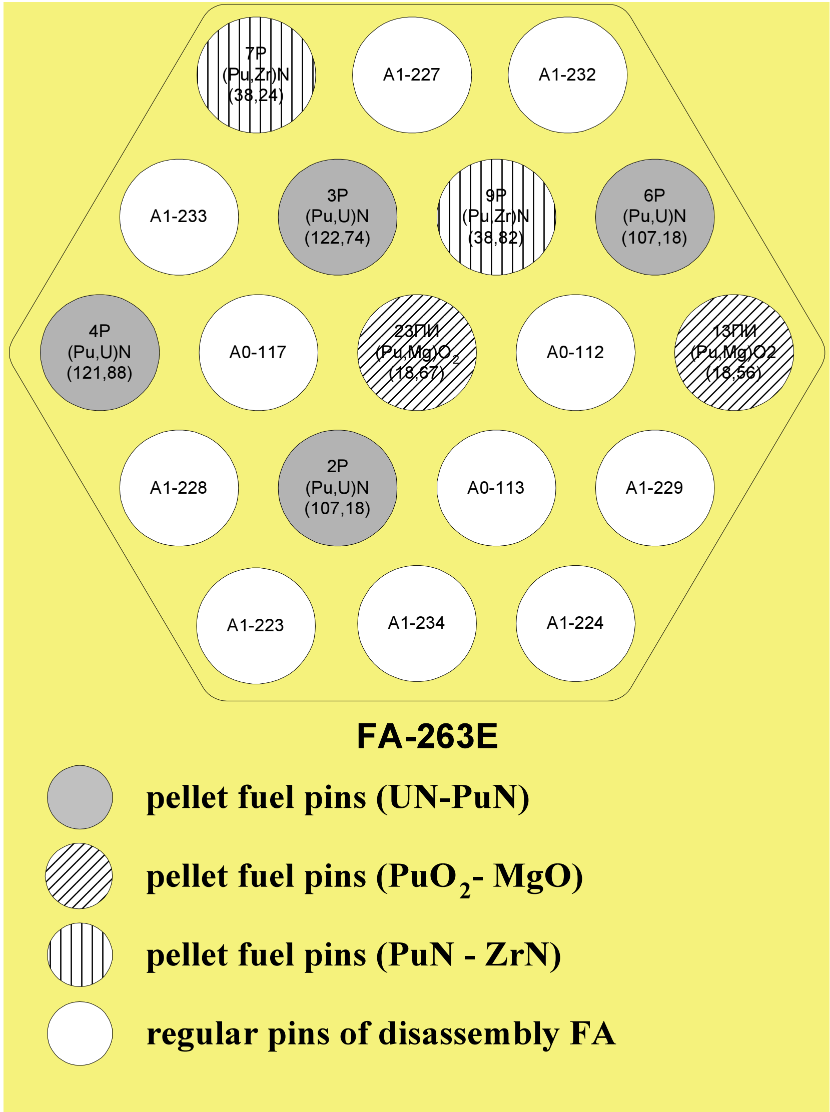
**completion of PIE – 2008.**

- **4** fuel pins with **$UPu_{0.45}O_2$ pellets** – VNIINM, Moscow
- **4** fuel pins with **$UPu_{0.45}O_2$ vibropac** fuel –RIAR, Dimitrovgrad
- **2** fuel pins with **$UPu_{0,45}N$ pellets** –VNIINM, Moscow
- **2** fuel pins with **$UPu_{0.6}N$ pellets** – VNIINM, Moscow
- **2** fuel pins with **PuN + ZrN pellets** –VNIINM, Moscow
- **2** fuel pins with **$PuO_2$ + MgO pellets** -IPPE, Obninsk

# BOR-60 irradiation device with removable fuel pins




1 - head
2 - spacer grid
3 – pins bundle
4 - wrapper
5 - throttle
6 - adapter
7 – tail



# BORA-BORA experiment: irradiation

Start of fuel irradiation in two dismountable FAs - August 2000. One FA - 4 pins with MOX pellets, 4 with MOX vibro, one FA - 4 nitride and 4 inert matrices fuel pins.

On November, 2002 FAs were unloaded for intermediate PIE at burn-ups 5.4 - 11.3 at% for different fuels (1-st irradiation stage).

After intermediate examination part of fuel pins was discharged for destructive PIE, the others were taken back for irradiation prolongation (2-nd irradiation stage) - December 2, 2003.

The irradiation has been completed on May 2005. All pins are intact.

PIE of $UPu_{0.45}N$ and $UPu_{0.6}N$ fuels are completed at end of 2008.

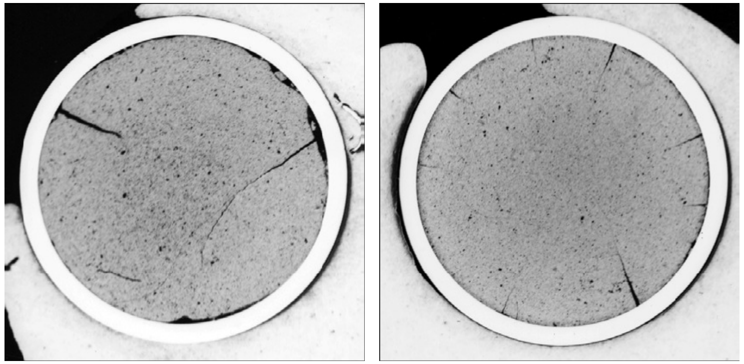
[a] *1-st irradiation stage / 2-nd stage*

| Fuel type | $UPu_{0.6}N$ | $UPu_{0.45}N$ |
|---|---|---|
| Irradiation time, efpd | 900/514[a] | 900/514[a] |
| Max burn-up, at. %. | 12.1/7.0[a] | 9.4/5.4[a] |
| Max dose, dpa | 43/23[a] | 43/23[a] |
| Max cladding temperature, $^0C$ | 604 | 567 |
| Max linear rating, kW/m, BOL/EOL | 54.5/35 | 41.9/26 |
| Max fuel temperature, $^0C$, BOL/EOL | 1760/1110 | 1490/930 |

# UPuN in BORA-BORA: PIE results

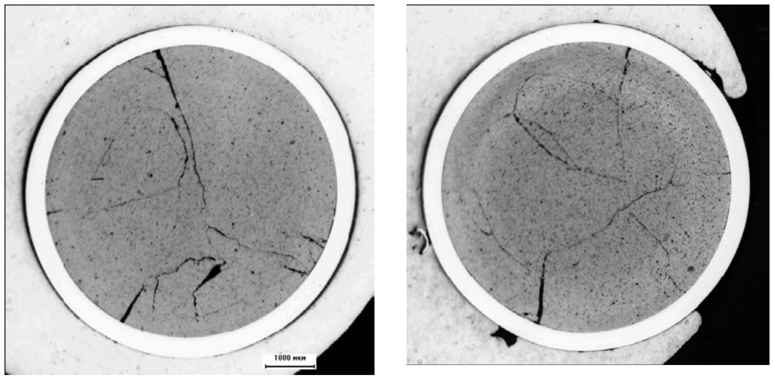
*$UPu_{0.45}N$ – core top (a), core middle (b),*

*a)* *b)*



*$UPu_{0.6}N$ – core top (c), core middle (d)*

c) d)




| Fuel type | $UPu_{0.6}N$ | $UPu_{0.45}N$ |
|---|---|---|
| Swelling rate, % / at% | (0,48–0,68) ±0,04% | (0,64–1,11) ±0,04% |
| Fission gas release, % | 19 | 19.3 |
| Max cladding corrosion depth, µm | no | 15 (at upper core section) |
| Some radial cracking, but no evident structure changing besides small intragranular porosity and coagulation of along border grain pores that confirms fuel temperature calculation correctness | | |
| No nitride dissociation was detected on microstructure and on nitrogen content changing. | | |

# UPuN in BORA-BORA: PIE results

- The positive results of irradiation tests of high-purity mixed nitride at BOR-60 reactor up to 12.1at% at maximum linear rating of 54,5 kW/m may be explained by high initial homogeneity of Pu distribution, low oxygen and carbon contents ( <0,15 w% and < 0,1 w%), uniform porosity distribution, combination of intra-grain and along grains borders pores.
- The performance of nitride fuel pin as well as oxide is limited by irradiation stability of cladding steel, and also by fuel-cladding mechanical interaction (BR-10 reactor experience).
- The positive BORA-BORA results confirm the possibility to provide at least 12at% of burn-up for He-bonded pins at initial fuel porosity increase. The BORA-BORA nitrides porosity is 15%.

# Principal results of fuels study: metal fuels

**Two techniques for alloyed metal fuel fabrication are under study in Russia: hot extrusion and injection casting . BR-10, BOR-60 fuel pins with U-Zr, U-Pu-Zr have been fabricated. One full scale FA with U-Pu-Zr has been irradiated at BOR-60 up to 10 at%. No PIE.**

- **As results of many years investigations in RIAR, Dimitrovgrad, the basic principles of radiation growth, swelling, gas release, corrosion behavior of unalloyed U, U-Pu with and without protective layers have been determined. Blanket, absorber, core fuel pins with U and U-Pu have been fabricated and irradiated at the BOR-60 and BN-350 reactors. The tests results have demonstrated the principal possibility to provide the performance of He-bonded pins with metal fuel of high smeared density ($\geq$12.5g/cm$^3$)**

- **Using of metal fuel in BN-type reactors is restricted by the necessity of sodium temperature and, accordingly, the reactor efficiency decreasing**

# Principal results of fuels study: BORA-BORA inert matrices fuels

***Transmutation and incineration are innovative options in management and disposal of fission products and actinides. In order to improve the efficiency of these processes, materials inert to neutron activation are of interest.***

- Irradiated **PuN-ZrN** has 2-phase structure consisting of solid solutions based on Zr and Pu nitrides differing by its proportion and fission products content.

  **Swelling rate <0,1%/1at.%.**
  **Gas release ~ 1%.**
  **Max cladding corrosion depth ~ 15μm.**
- Irradiated **PuO2 – MgO** has 2-phase structure consisting of Pu and Mg dioxides with fission products dissolved .

  **Swelling rate ~ 0.5%/1at%.**
  **Gas release - 9%.**
  **Max cladding corrosion depth < 10μm.**

| Fuel type | $PuO_2$+MgO | PuZrN |
|---|---|---|
| Pu content, % | 35.8 | 37.5 |
| Density, %th | 88…91 | 83…84 |
| Irradiation time, efpd | 900/514[a)] | 900/514[a)] |
| Max burn-up, at. %. | 19.0/11.1[a)] | 19.4/11.3[a)] |
| Max cladding temperature, $^0$C | 551 | 547 |
| Max linear rating, kW/m, BOL/EOL | 10.0/8.8 | 20.7/18.2 |

[a)] *end of 1-st irradiation stage / end of 2-nd stage*

# Inert matrices fuels: out-of-pile properties

**ISTC Project #2680 "MATINE - Study of Minor Actinide Transmutation in Nitrides: modeling and measurements of out-of-pile properties"**

- **(Pu,Zr)N** samples with **ZrN=60mol% of 85%th, 93%th density done** by direct nitridation of metals. Powder mixing by VR process , solid solution has received .
- Study of (PuZr)N properties :

**High-temperature creep** by uni-axial compression under very-high-purity argon. At steady-state phase the creep rate temperature dependence is satisfactorily described by Arrhenius equation. Creep rate increases linearly with increasing load.

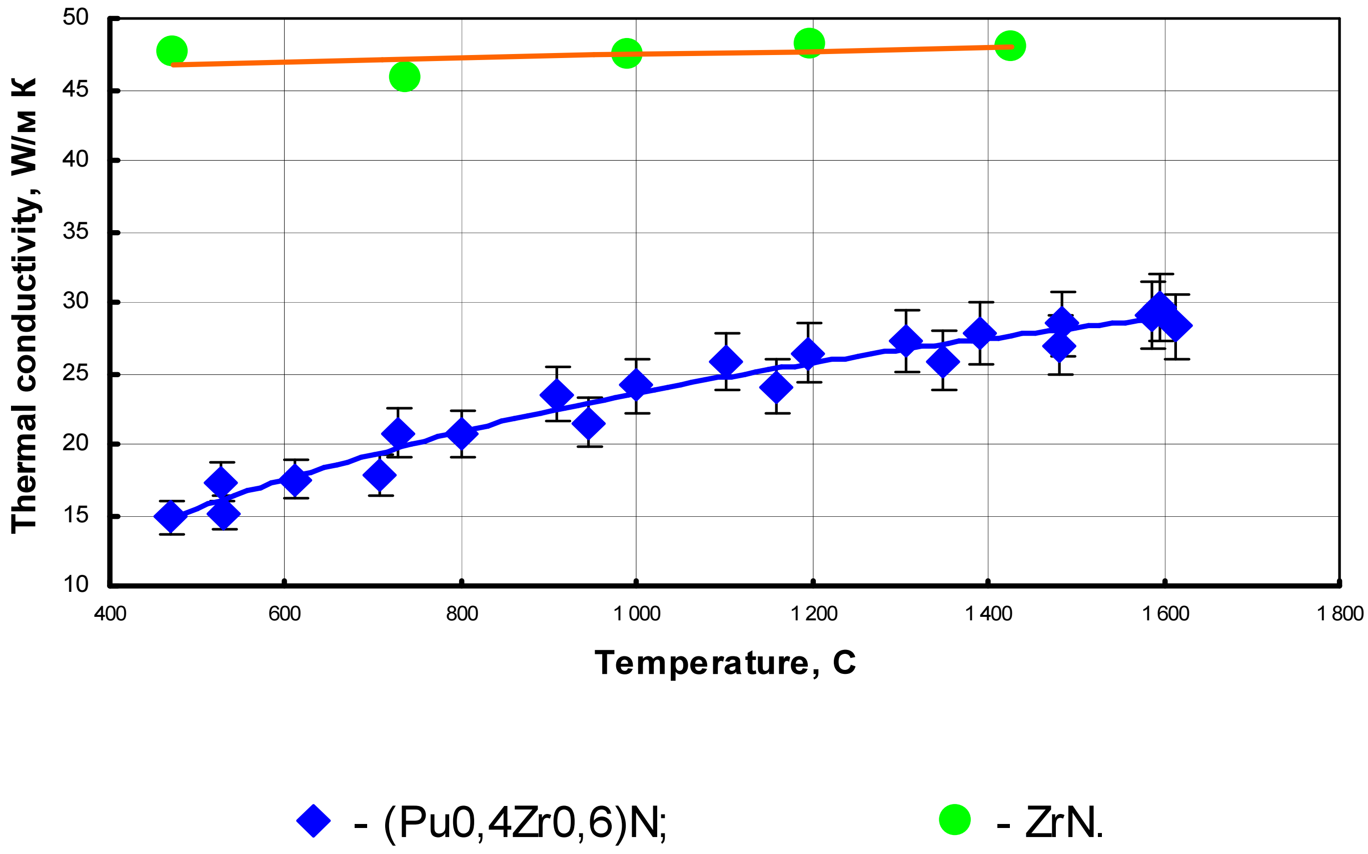
**High-temperature stability** study revealed differences in (PuZr)N behavior at 2200°C - 2300°C in various environments.

- In vacuum fuel stability was the lowest. Presumably, Pu evaporated through matrix open porosity.
- In argon and nitrogen the samples revealed high stability; however separation of phases with high plutonium concentration was observed.

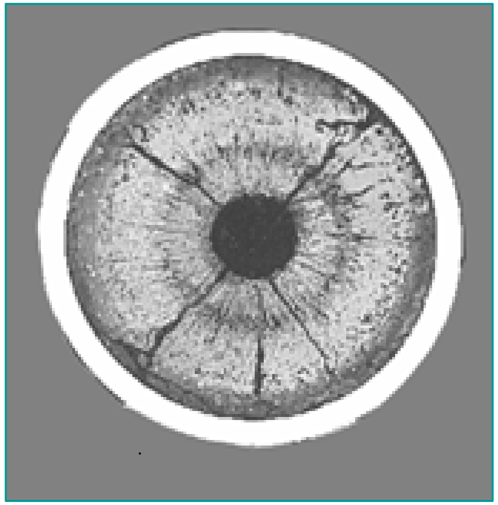
**Thermal expansion** of ZrN and (Pu,Zr)N samples was investigated by using the dilatometric system at temperatures of 20ºC to 1400ºC in argon + hydrogen . For ZrN good agreement with literature data .

**Thermal conductivity** from 500°C to 1600°C was studied by laser flash method under vacuum. Certain discrepancies with respective foreign data, that can be explained by different physico-chemical and structural samples properties due to the difference of fabrication processes.

# MATINE: (PuZr)N and ZrN thermal conductivity, 100% density



# Principal results of fuels study: MA fuels



$(UNp)O_2$ after BOR-60 irradiation up to ~20 at %

- **Russian-French AMBOINE experiment (AMericium in BOr-60: INcineration Experiment): possibility of Am recycling by pyrochemical methods has been studied. The program included fabrication of BOR-60 type fuel pin with vipac $(UAm)O_2$ in the core and with vipac $UAmO_2$+MgO in axial blankets. Investigations on Am/REE (rare earth elements) and MgO separation by selective precipitation in molten salts have been carried out also.**
- **DOVITA program: irradiation of $UNpO_2$ vi-pack fuel to 20at% has been done at BOR-60. No principal difference is seen comparing MOX or uranium oxide .**
- **ISTC MATINE Project: the possibility has been considered on fabrication of (Pu,Am,Cm,Zr)N with up to 10mol% Cm on RIAR site. Electrolytic refining in molten chlorides on liquid metal cathode was proposed as main flow sheet. The technical-economical estimations have shown the technical feasibility of the offered processes at RIAR site.**

# Fuel codes development

- The principle code for the performance evaluation of cylindrical fuel pins is the KONDOR code made in IPPE, Obninsk . On the base of this code the KORAT code has been developed in VNIINM, Moscow, for MOX pellet fuel pins calculation and the VIKOND code in RIAR, Dimitrovgrad, for vibro MOX fuel pin calculations. All these codes realize the calculation of thermo-mechanical characteristics of one separate axial section of a fuel pin.

- With the purpose of computer resources consumption minimization and calculation efficiency increase the DRAKON-3D code has been developed in IPPE. KONDOR code modules are the key ones in DRAKON code structure. DRAKON code may be used for temperature and stress-strain state calculations of cylindrical fuel pins with MOX pellet and different types of dense fuels both in steady conditions and transients. It considers Nz ~100-200 pin axial sections.

# Fuel codes development :DRAKON code

**Two following models have been developed for swelling calculation of dense fuel types :**

- ***Model of spherical cells.*** **The fuel is considered as dispersion composition. It is supposed, that dispersion fuel consists of the identical cells, regularly located in fuel volume (according to face centered cube scheme). Each cell represents a thick-walled spherical cladding made of non-fissile (for example, ZrN) material with spherical grain of fissile material inside.**

- ***Model of spherical gas pores*****. According to this model, the volume of fuel is conditionally divided into regular spherical cells, each of which contains one pore**. ***(AnZrN (An - Pu, Am, Cm, Np) is considered as homogeneous mechanical mixture of ZrN matrix and fissile material).***

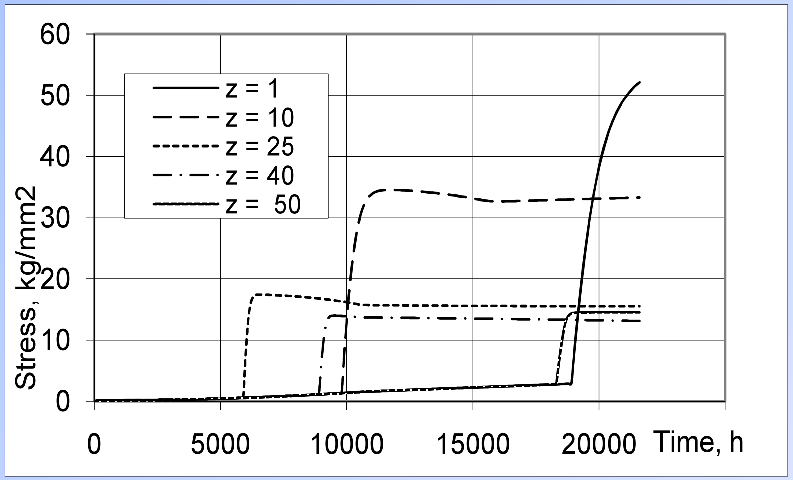
***For UPuN swelling calculation the model of spherical gas pores is used. The verification of the model has been done using experimental data for nitride fuel pins irradiated in various reactors. The required coefficients have been received.***

# Fuel codes development: DRAKON code

For performance evaluation of fuel pins under conditions of so called "rigid" loading scheme typical for dense fuels the correct data on fuel swelling and creep are of principal importance as well as cladding deformation capability.

BORA-BORA results of four UPuN pins PIE data are important for the verification and modification of some code modules (temperature, fuel swelling and gas release).

An example of hoop stress calculation for inner surface of $UPu_{0.6}N$ cladding is in Figure. The first fuel-cladding contact is in middle axial sections wihere max fuel swelling rate. Stresses in "hot" sections relax with time, and max stress is realized in bottom sections with relatively "cold" fuel and absence of cladding high temperature creep




*Hoop stress per time in different axial section (z) of $UPu_{0.6}N$ pin*

# Conclusion

- **Different fuels: $PuO_2$, $UO_2$ (pellet, vibro), $UPuO_2$(pellet, vibro), UC, UN, UPuC, UPuN, oxide, nitride and carbide inert-matrices fuels, alloyed and non-alloyed metallic fuels have been studied in Russian SFRs (BN reactors). Recently the experiments with UPuN and MgO, ZrN –based fuels have been completed in BOR-60.**

- **In order to meet the requirements to nuclear installations of 4-th generation: sustainability, proliferation resistance, waste management, safety, economics, the R&D on commercial BN-K reactor are currently under way in Russia. As reference fuel MOX fuel is considered, as more long-term option nitride is assumed to be. Other types of dense fuels are planned to be studied as well.**

# Thank you for your attention!