2003 year

- Nuclear Power Development Strategy in Russia in the first half of XXI century
  - May 2000
- The initiative of President of Russia at the UN Millennium Summit
  - September 2000
- «Energy strategy of Russia for the period up to 2020»
  - May 2003

Federated Programme
- «Energy cost-effective economics»
  - 2003

Russian Federation Government Decree
- General directions of electric power system reformation
Closing of the nuclear fuel cycle is a strategic line of nuclear power development in Russia, which will ensure more efficient use of natural nuclear fuel and artificial fissile materials produced by reactors (e.g., plutonium), will minimise radwaste from fuel reprocessing and will help approach a radiation equivalence between the buried waste and the mined natural fuel.
I. Up to 2007/2010:

- Dry storage of SF of TR (VVR and RBMK)
- RT – 1 Plant (“Mayak”) – semi – closed FC:
  - reprocessing of HEUF of ER/SM
  - reprocessing of LEUF of WWR-440
  - Production of LEUF for RBMK Pu storage
- R&D for new technologies:
  - TR fuel reprocessing (RT-2, Krasnoyarsk)
  - FR (BREST) fuel (U-P/N) reprocessing
  - FR/TR fuel (U-Pu) production
  - transmutation (in FR)
- Design and construction of BREST – 300
- System studies of nuclear power development in Russia in 21 century

III. After 2010 (up to ~ 2030)

- RT – 2 plant construction, start of processing of TR fuel
- Fuel supply for FRs (BREST) and reprocessing of spent fuel
- Approach to radiation – equivalent wast disposal
- Minor actinide transmutation
MODEL of IDEAL NUCLEAR FUEL CYCLE

NUCLEAR FUEL → SPENT NUCLEAR FUEL → MA Fusion Products (Sr, Cs, Tc, I)

GEOLOGICAL FORMATIONS

Gr = Gradwaste
Principle of radiation equivalency

- Equivalency of radio-toxicity from LL HLRW at ultimate disposal and used uranium radio-toxicity;
- Matching to global radiation equilibrium;
- Need of a local approach;
- Need of considering technologic factors, as well as chemical and geochemical properties of radionuclides during long-term storage/disposal;
- Partitioning: decay time, geochemically alike behaviour;
- Immobilization in matrices exhibiting nature-alike stability.

Consequences - *Formulating qualitative requirements.*
PROBLEMS of IDEAL NFC

99,9% MA (Np, Am, Cm)
99% Sr, Cs, Tc, I

- Fractioning of SNF reprocessing products
- Transmutation of MA, Tc, I
- Storage of Cs, Sr
<table>
<thead>
<tr>
<th>STORAGE</th>
<th>REPROCESSING</th>
</tr>
</thead>
<tbody>
<tr>
<td>WET</td>
<td>without fractionating</td>
</tr>
<tr>
<td></td>
<td>too much LLW</td>
</tr>
<tr>
<td></td>
<td>Small scale reprocessing</td>
</tr>
<tr>
<td>DRY</td>
<td>Fractionation</td>
</tr>
<tr>
<td>(URC)</td>
<td>Postponing of the bulk of SNF reprocessing</td>
</tr>
<tr>
<td>LNPP</td>
<td>Modernisation</td>
</tr>
<tr>
<td>Bi l NPP</td>
<td>RT-1 Reprocessing</td>
</tr>
</tbody>
</table>

**Pespectives:**
- Modernisation of RT-1 Reprocessing technology choice (RT-2)
Principal flowsheet of the extraction cycle with simultaneous RW fractionating
Major shortcomings of the present system of low-level liquid RW handling at radiochemical and chemical engineering facilities of the PA «Mayak»

- Large volumes of initial liquid RW (up to 700,000 m³/year);
- Discharge of radioactive secondary wastes in the form of solutions and pulps to the open basin of the Techa cascade that is ecologically unacceptable;
- Large volumes of secondary wastes (up to 20% of the initial volume of liquid RW);
- Application of technological procedures resulting in a significant salinization of initial liquid RW (for example, nearly twice as much salts are added than extracted in the process of ion-exchange purification);
- Insufficient purification from some nuclides;
- Discharge of laundry waters to the reservoir B-2 without any their purification.
Major directions of improving the present system of low-level liquid RW handling at the PA «Mayak»

- Reduction of volumes and salinization of produced liquid RW by way of replacing the precipitating and sorption procedures by low-reagent techniques (membrane ones, etc.). Introduction of a closed water cycle, if possible.

- More effective purification, as well as conditioning of liquid RW concentrates with a full ceasing the discharge of radioactive waters and pulps to open basins (storages.)

- Making optimal decisions concerning the containerization, long-term storage and disposal of conditioned RW.

**State of affairs**

*The PA "Mayak", in cooperation with the VNIINM and other institutions, elaborated technological approaches as concerns the modernization of the present techniques and elimination of the above-said shortcomings.*

*The proposed flowsheets are based on low-reagent membrane and sorption technologies and conditioning of the secondary wastes by methods of cementing and vitrification using ceramic and induction melters ("cold" crucible).*

*In the nearest future, it is planned to specify necessary equipment, and to begin
Promising membrane and sorption flowsheet of low-level reprocessing at the PA “Mayak”

- **Ozonization**
  - Solutions of the special sewer system 470,000 m³
  - Plant 235, Plant 23, Plant 45, Plant 156

- **Ultrafiltration**
  - Filtrate
  - Concentrate 4,000 m³

- **Back osmosis**
  - Filtrate
  - Concentrate 20,000 m³

- **Ion-exchange**
  - Purified water to be discharged to the open drainage system or reused 444,000 m³

- **Collecting and neutralizing**
  - Transportation to Plant 235
  - Cementing

- **Desalinated water B-1**
  - 400,000 m³

- **Solutions of the laundry**
  - 70,000 m³

- **Ferrocyanide**

- **Concentrate**
  - 4,000 m³

- **Concentrate**
  - 20,000 m³

- **Regenerates and industrial waters**
  - 2,000 m³
NPP spent fuel reprocessing with the use of Purex process
NPP spent fuel reprocessing with the use of Syper Purex process.
NPP spent fuel combined reprocessing
NPP spent fuel REPA-process
Projected flowsheet for NPP spent fuel reprocessing at RT-2 plant.

- WWER-1000 storage
- Assembly shearing
- Dissolution
- $I_2$
- Fast reactor assemblies
- Storage and enrichment
- HLW partitioning
- Modified Purex process
- FP
- REE
- Cs, Sr
- Cm
- Am
- Pu
- $U_{depl.}$
- Np
- Tc
- U
- Pu
Flow diagram for promising NPP spent fuel reprocessing with the use of Syper Purex process.
NPP spent fuel combined reprocessing flow diagram.

- WWER-1000 storage
- Assembly destruction
- Fuel oxidation
- Fluorination
- Gaseous phase 95% U, FP (a part)
- Sorption
- Distillation
- Tc
- U
- Enrichment
- HLW partitioning
- Extraction process
- FP
- REE
- Cs, Sr
- Np
- Am
- Pu, U
- Fast reactor assemblies
Basic flowsheet of INF reprocessing

Spent nuclear fuel

Thermomechanical destruction of shell

Gas conversion of uranium oxide into nitrate

Correction of gas mixture

Separation of RNG and tritium

Leaching of HUN melt

Uranium formation and purification by way of crystallization of HUN melt

Extractive reprocessing with simultaneous fractioning

CO$_2$N$_2$O$_4$

Shell material

HUN crystals

Mother liquid

REE/TPE and FP fraction

U/Pu (U+Pu)

Cs/Sr and FP fraction
Comparison of reprocessing technologies

<table>
<thead>
<tr>
<th>Technology</th>
<th>Distinguishing features</th>
<th>Advantages</th>
<th>Limitations</th>
</tr>
</thead>
<tbody>
<tr>
<td>PUREX-process</td>
<td>One process line for uranium purification</td>
<td>Industrial experience is available</td>
<td>A large volume of evaporation operations</td>
</tr>
<tr>
<td>Combined flowsheet</td>
<td>Separation of 95% of uranium in the form of UF₆</td>
<td>Uranium arrives directly at enrichment stage</td>
<td>A large volume of F-radwaste, two complicated technologies</td>
</tr>
<tr>
<td>REPA-process</td>
<td>Crystallization of 95% of uranium in the form of UO₂(NO₃)₂</td>
<td>Simplified procedure of the main part of uranium</td>
<td>There is no practical experience of operation in radiochemical industry</td>
</tr>
<tr>
<td>Super Purex</td>
<td>Single partitioning cycle</td>
<td>Industrial experience is available</td>
<td>A large volume of evaporation operation</td>
</tr>
</tbody>
</table>
TRANSMUTATION NUCLEAR FUEL CYCLE

NUCLEAR FUEL

SPENT NUCLEAR FUEL

0.1%MA Fusion Products 1%(Sr, Cs, Tc, I)

GEOLOGICAL FORMATIONS

Gtc = Gradwaste
Transmutation nuclear fuel cycle

Depleted uranium

Manufacturing of FAs and targets for transmutation

Fast reactor

Reprocessing of spent fuel from fast reactor, Cm storage (50-100 years)

Thermal reactor

Manufacturing of Fas for TR

Enrichment

U mining

Reprocessing of spent fuel from thermal reactor

0.1% U, Pu, MA, 1% Sr, Cs, Tc, I, 100% Th, Pa, Bk, Cf, other FP

0.1% U, Pu, MA, 1% Sr, Cs, Tc, I, 100% Th, Pa, Bk, Cf, other FP

Temporary RAW storage

Using and storage of Sr and Cs

RAW immobilization

Earth

Radiotoxicity of used uranium = Radiotoxicity of disposed RAW
Vibropacking technique

Fuel rods with granulated fuel are fabricated by vibropacking technique according to the standard procedure (in glove boxes or shielded cells) that has been used at RIAR for 20 years.

The main advantages of the vibropacking technique and vibropacked fuel rods are the following:

- Simplicity and reliability of the production process due to the reduced number of process and control operations, that makes the automation and remote control of the process easier.
- Possibility of usage of the granulate in any form; both in the form of a homogeneous composition and mechanical mixture.
- Reduced thermo mechanical impact of vibropacked fuel on the cladding (as compared with a pelletized core).
- More flexible requirements for the inner diameter of the fuel rod claddings.
Production and testing of vibropacked fuel rods on the basis of (U, Pu)O₂

<table>
<thead>
<tr>
<th>Fuel type</th>
<th>Number of fuel assemblies</th>
<th>Burnur, max.%</th>
<th>Load, kW/m</th>
<th>Temperature, °C</th>
<th>Reactor</th>
</tr>
</thead>
<tbody>
<tr>
<td>(U, Pu)O₂</td>
<td>low-background, high-background</td>
<td>330</td>
<td>30,3</td>
<td>51,5</td>
<td>720</td>
</tr>
<tr>
<td>UO₂ + PuO₂</td>
<td>low-background, high-background</td>
<td>132</td>
<td>14,8</td>
<td>45</td>
<td>705</td>
</tr>
<tr>
<td>(U, Pu)O₂</td>
<td>low-background</td>
<td>9+(3)</td>
<td>11,1</td>
<td>46</td>
<td>680</td>
</tr>
<tr>
<td>(U, Pu)O₂</td>
<td>high-background</td>
<td>4</td>
<td></td>
<td>development of the production technique</td>
<td>BN-600</td>
</tr>
</tbody>
</table>
Experience in introduction of minor-actinides into the fuel composition and irradiation results

Within the framework of the DOVITA program a batch of granulated uranium-neptunium oxide fuel with neptunium content of 5 per cent was produced.

On the basis of the fuel produced, vibropacked fuel rods with a fast reactor cladding were fabricated using the equipment designed for BOR-60 fuel rods fabrication.

The isotopic content of fuel rods with neptunium fuel (irradiation during 694 days in BOR-60 reactor)

<table>
<thead>
<tr>
<th>Isotope</th>
<th>Mass, g</th>
<th>Changes in isotope mass</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Before irradiation</td>
<td>After irradiation</td>
</tr>
<tr>
<td>$^{235}\text{U}$</td>
<td>53.5</td>
<td>42.9</td>
</tr>
<tr>
<td>$^{238}\text{U}$</td>
<td>16.5</td>
<td>15.96</td>
</tr>
<tr>
<td>$^{237}\text{Np}$</td>
<td>4.63</td>
<td>3.93</td>
</tr>
<tr>
<td>$^{234}\text{U}$</td>
<td>-</td>
<td>0.012</td>
</tr>
<tr>
<td>$^{236}\text{U}$</td>
<td>-</td>
<td>1.69</td>
</tr>
<tr>
<td>$^{238}\text{Pu}$</td>
<td>-</td>
<td>0.34</td>
</tr>
<tr>
<td>$^{239}\text{Pu}$</td>
<td>-</td>
<td>0.31</td>
</tr>
<tr>
<td>$^{240}\text{Pu}$</td>
<td>-</td>
<td>$4.2*10^{-3}$</td>
</tr>
<tr>
<td>FP</td>
<td>-</td>
<td>9.48</td>
</tr>
</tbody>
</table>

At present, 20 per cent of the burnup is achieved in the fuel rods with neptunium fuel.

Experimental fuel rods with $(U,\text{Pu},\text{Am})\text{O}_2$ fuel, containing 3 per cent of americium, have been prepared, and in the year 2002 they will be subjected to irradiation in the BOR-60 reactor.
## Properties of oxide and nitride fuel

<table>
<thead>
<tr>
<th>Properties</th>
<th>UPuO$_2$</th>
<th>UPuN</th>
</tr>
</thead>
<tbody>
<tr>
<td>Density (g/cm$^3$)</td>
<td>11,05</td>
<td>14,32</td>
</tr>
<tr>
<td>Contains of fissile materials in 1 cm$^3$ fuel (g/cm$^3$)</td>
<td>9,74</td>
<td>13,53</td>
</tr>
<tr>
<td>Thermal conductivity by temperature 500-1000°C (wt/mK)</td>
<td>2,2-2,0</td>
<td>20-22</td>
</tr>
<tr>
<td>Temperature of melting (K)</td>
<td>2950</td>
<td>3050</td>
</tr>
<tr>
<td>Interaction with thermal carriers: Na, Pb, Pb-Bi</td>
<td>Interaction with Na and (Na$_3$UPuO$_4$+Q) creation</td>
<td>Not interaction</td>
</tr>
</tbody>
</table>
Studies are under way to investigate the possibilities of keeping uranium and plutonium inseparable and of satisfying the requirements concerning fuel purification and waste fractioning (see below) afforded by various radiochemical techniques, such as:

- aqueous;
- molten chloride electrolysis, with actinide reduction into metals or nitrides;
- metallurgical refining, with no nitride breakup in any reprocessing stage;
- in molten fluorides;
- gas fluorination;
- electrolysis of molten fluorides;
- recrystallization in molten molybdates and phosphates, etc.

The basic process routes, equipment mix, etc. have already been worked out. The developers are now reviewing the requirements for radiochemical techniques, estimating the costs and investigating the technical feasibility of the project.
NPP-attached reprocessing plant
for regeneration and production of the BREST reactor fuel

Concept of the station-attached Plant for regeneration and production of the BREST reactor fuel:

- Production of mononitride fuel from the BREST spent fuel at the stage of pyrochemical reprocessing
- Production of mononitride fuel pellets
- Fabrication of fuel rods with sublayer on the basis of pelletized fuel
- Manufacturing of the BREST fuel assemblies

Diagram:
- Ar, Ar
- U, Pu
- Cd
- Spent fuel
- Anodic basket
- Cadmium cathode
- \(? = 450 \degree C \ LiCl - KCl\)
Experience in reprocessing of spent fuel of the BOR-60 and BN-600 reactors

<table>
<thead>
<tr>
<th>Fuel type</th>
<th>Burnup, %</th>
<th>Mass, kg</th>
<th>Period</th>
<th>Reactor</th>
</tr>
</thead>
<tbody>
<tr>
<td>$\text{UO}_2$</td>
<td>7.7</td>
<td>2.5</td>
<td>1972..1973</td>
<td>BN-350</td>
</tr>
<tr>
<td>$(\text{U},\text{Pu})\text{O}_2$</td>
<td>4.7</td>
<td>4.1</td>
<td>1991</td>
<td>BOR-60</td>
</tr>
<tr>
<td>$(\text{U},\text{Pu})\text{O}_2$</td>
<td>21..24</td>
<td>3.5</td>
<td>1995</td>
<td>BOR-60</td>
</tr>
<tr>
<td>$\text{UO}_2$</td>
<td>10</td>
<td>5</td>
<td>2000</td>
<td>BOR-60</td>
</tr>
<tr>
<td>$(\text{U},\text{Pu})\text{O}_2$</td>
<td>10</td>
<td>12</td>
<td>20002001</td>
<td>BOR-60</td>
</tr>
</tbody>
</table>

PuO$_2$, UO$_2$ and MOX Decontamination factors (DF) from main FPs

<table>
<thead>
<tr>
<th>Fuel type</th>
<th>Main FPs</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Ru- Rh</td>
</tr>
<tr>
<td>$\text{PuO}_2$ for BN-350 (test, 1991)</td>
<td>50</td>
</tr>
<tr>
<td>$\text{PuO}_2$ for BOR-60 (test, 1995)</td>
<td>33</td>
</tr>
<tr>
<td>$\text{UO}_2$ for BOR-60 (test, 2000)</td>
<td></td>
</tr>
<tr>
<td>$(\text{U},\text{Pu})\text{O}_2$ for BOR-60 (test, 2001)</td>
<td></td>
</tr>
</tbody>
</table>
## STATUS and PERSPECTIVES

<table>
<thead>
<tr>
<th>RW type</th>
<th>Status</th>
<th>Perspectives</th>
</tr>
</thead>
<tbody>
<tr>
<td>HLW</td>
<td>vetrification ceramisation</td>
<td>final disposal</td>
</tr>
<tr>
<td>MLW</td>
<td>vetrification ceramisation</td>
<td>final disposal</td>
</tr>
<tr>
<td>LLW</td>
<td>final disposal too much</td>
<td>decreasing</td>
</tr>
</tbody>
</table>
Experience in disposition of radioactive waste arising from pyrochemical regeneration of spent fuel

Vitrification of HLW resulted from pyrochemical process

<table>
<thead>
<tr>
<th>Characteristics</th>
<th>Phosphate deposit</th>
<th>Spent salt electrolyte</th>
<th>Phosphate deposit + spent salt electrolyte</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Pb(PO₃)₂, NaPO₃</td>
<td>NaPO₃, AlF₃, Al₂O₃</td>
<td>NaPO₃, AlF₃, Al₂O₃</td>
</tr>
<tr>
<td>Type of glass matrix</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>glass matrix</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Quantity of wastes introduced, %</td>
<td>28</td>
<td>20</td>
<td>36</td>
</tr>
<tr>
<td>Leaching rate of ¹³⁷Cs on 7-th day, g/cm² * day</td>
<td>7*10⁻⁶</td>
<td>7*10⁻⁶</td>
<td>4*10⁻⁶</td>
</tr>
<tr>
<td>Thermal stability, °C</td>
<td>400</td>
<td>400</td>
<td>400</td>
</tr>
<tr>
<td>Radiation resistance</td>
<td>10⁷ Gy (for γ and β)</td>
<td>10¹⁸ α-decay/g</td>
<td></td>
</tr>
</tbody>
</table>
Experience in disposition of radioactive waste arising from pyrochemical regeneration of spent fuel

Ceramization of HLW arising from pyrochemical process

<table>
<thead>
<tr>
<th>Characteristics</th>
<th>Type of high-level wastes</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Phosphate deposit</td>
</tr>
<tr>
<td>Type of ceramics</td>
<td>monazite</td>
</tr>
<tr>
<td>Method of introduction into ceramics</td>
<td>pressing, calcination, ≤850°?</td>
</tr>
<tr>
<td>Quantity of waste introduced into ceramics, %</td>
<td>100</td>
</tr>
<tr>
<td>Leaching rate of $^{137}$Cs on 7-th day, g/cm2 * day</td>
<td>$1 \times 10^{-6}$</td>
</tr>
<tr>
<td>Thermal stability, °C</td>
<td>850</td>
</tr>
<tr>
<td>Radiation resistance</td>
<td>$5 \times 10^8$ Gy (for $\gamma$ and $\beta$)</td>
</tr>
</tbody>
</table>
PROBLEMS

Requirements

Closed nuclear fuel cycle
Optimization of technological system (NPP-FC)
Maximum level of inherent safety
Minimum quantity of wastes (radiation equivalent waste disposal)
Minimum fuel cycle costs, maximum efficient use of natural resources
Minimum risk of fissile materials proliferation
Sustainable development of nuclear power - military and political neutrality of the industry: from the concept of «political control over nuclear power» to the concept of «politically neutral nuclear power»

technological support of nonproliferation

• no Pu production in blankets
• no Pu extraction during spent fuel reprocessing
• U enrichment to be given up
Conclusions

1. CNFC is a strategic line in nuclear power development.

2. Large scale nuclear nuclear - nuclear power with fast reactors

3. CNFC for large scale nuclear power - transmutational NFC

4. CNFC problems – good subject for international cooperation.