NUCLEAR WASTE BURNER FOR MINOR ACTINIDES ELIMINATION

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NUCLEAR WASTE BURNER (NWB) – AN ADS INDUSTRIAL PROTOTYPE FOR MINOR ACTINIDES ELIMINATION

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n_TOF Winter School on Astrophysics, ADS, and First Results
Les Houches, France, February 24 -28, 2003

2003

ACTINIDES AND FISSION PRODUCTS

- ♦ Over 300 thousand tons of spent nuclear fuel (SNF) will be accumulated in technological lines of nuclear industry by 2010.
- ♦ The most hazardous isotopes in spent fuel will be:
  - Minor actinides:

    3000 t Pu; 140 t Np-237; 120 t Am-241, etc.
  - High-level fission products:

    250 t Tc-99; 90 t Cs-135; 60 t I-129, etc.
- ♦ Hundreds tons of high-level radioactive products are accumulated in the process of nuclear weapons development
- ♦ Total long-lived (>10^5 years) radioactivity exceeds $5 \cdot 10^8$ Ci

The main challenge for safe development of nuclear power is management of extreme quantities of radioactive materials

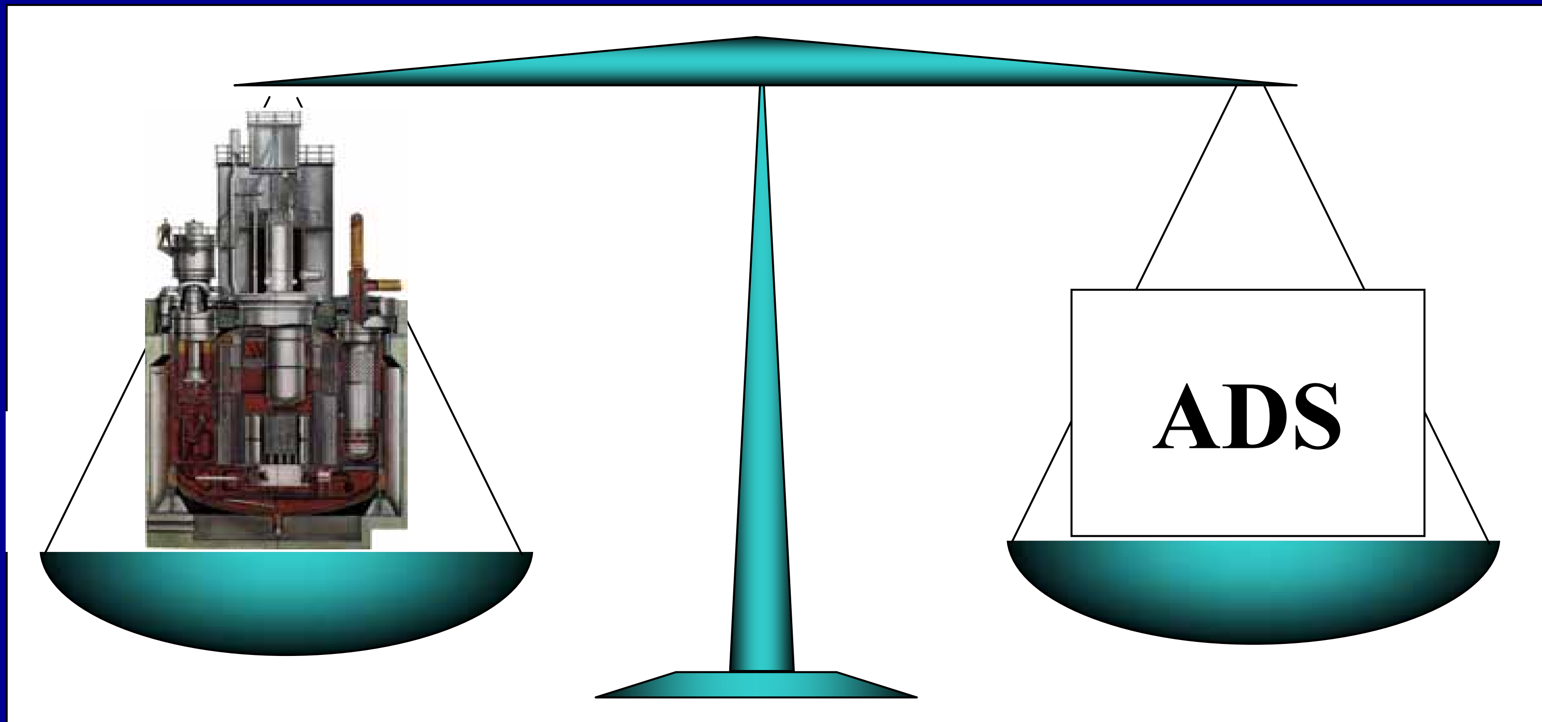
WAYS OF MANAGEMENT

Conservative - use of underground repository in deep geologic formations for the period over 100 000 years

▪

Dynamic - reprocessing of spent nuclear fuel, incineration of long-lived minor actinides and some fission fragments to decrease total long-lived radioactivity to the level, that allows a subsequent final disposal

FAST REACTORS AND ADS – SYSTEMS



Technologies brought to wide use Minimum auxiliary power production needed	Excluding the reactivity accidents Superfast neutrons More flexibility with fuel and MA loading

OBJECTIVE OF THE NWB - PROJECT

Creation of the world's first industrial accelerator driven system and demonstration of semi-industrial technology for safe transmutation of minor actinides and other high level wastes of nuclear power

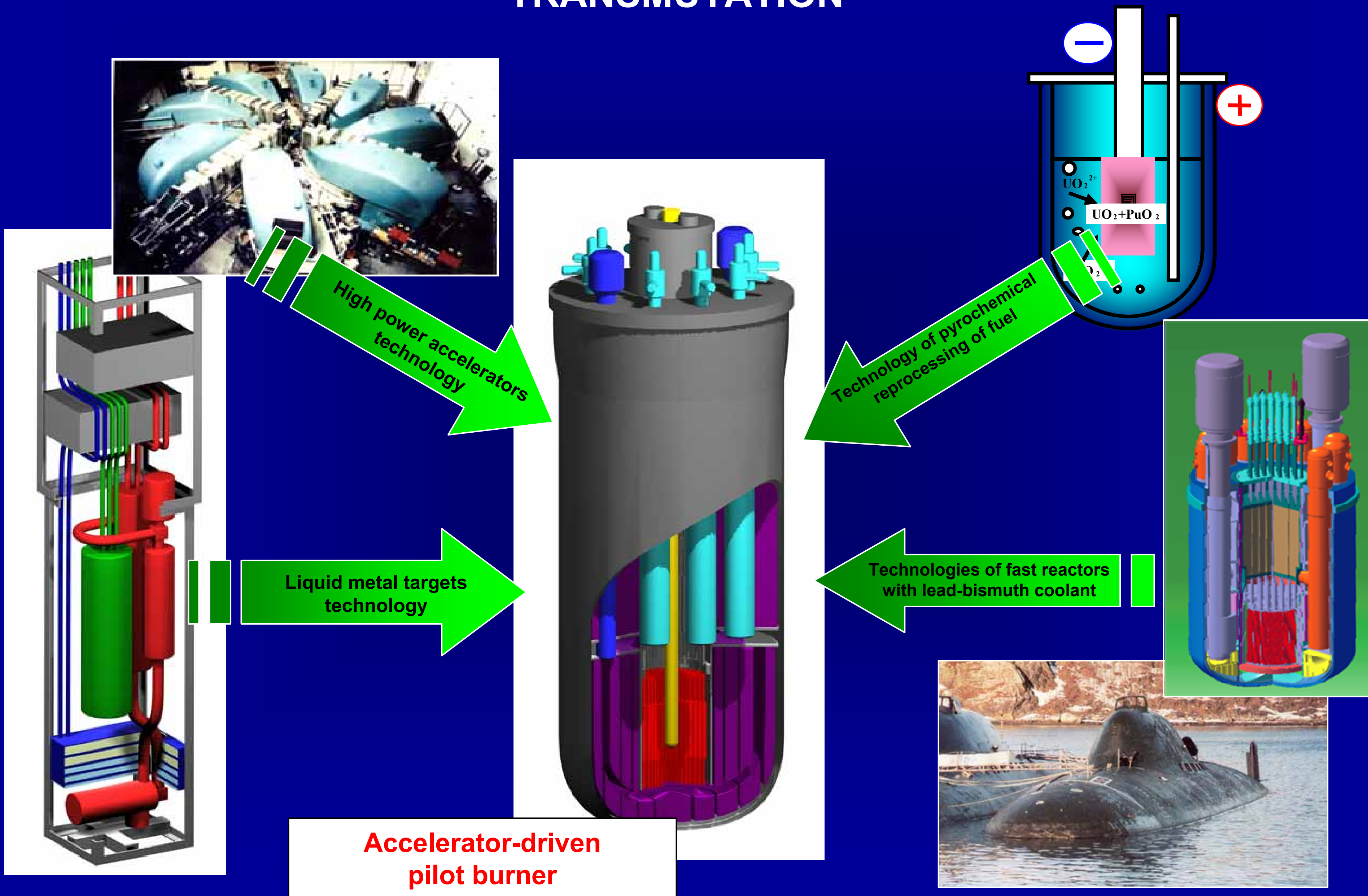
THE MAIN TASKS OF THE NWB - PROJECT

- Comprehensive demonstration the accelerator–driven system technology

- Gain the operational and technological experience for the later use in other projects

- Demonstration the technology of safe transmutation of minor actinides and other radioactive waste of the nuclear fuel cycle.

KEY COMPONENTS OF TECHNOLOGY FOR ACCELERATOR-DRIVEN TRANSMUTATION



TECHNOLOGY PREPAREDNESS

TECHNOLOGY	CURRENT STATUS
Fast neutron reactors	Have been used in industry of Russia, France, Japan and US
Heavy metal-cooled reactors and targets	Experience of Russia – 80 reactor years of nuclear power systems for submarines; R&D results for nuclear power designs in Russia, Europe, Japan and US
Technologies for pyrochemical reprocessing of SNF and remotely-controlled fabrication of fuel elements	Semi-industrial vibropack MOX fuel production for fast reactors in Russia
Technology of high-current proton accelerators	Operating accelerators in Europe, US, Japan and Russia

INSTITUTIONS READY TO WORK FOR NWB PROJECT WITHIN AN INTERNATIONAL COOPERATION

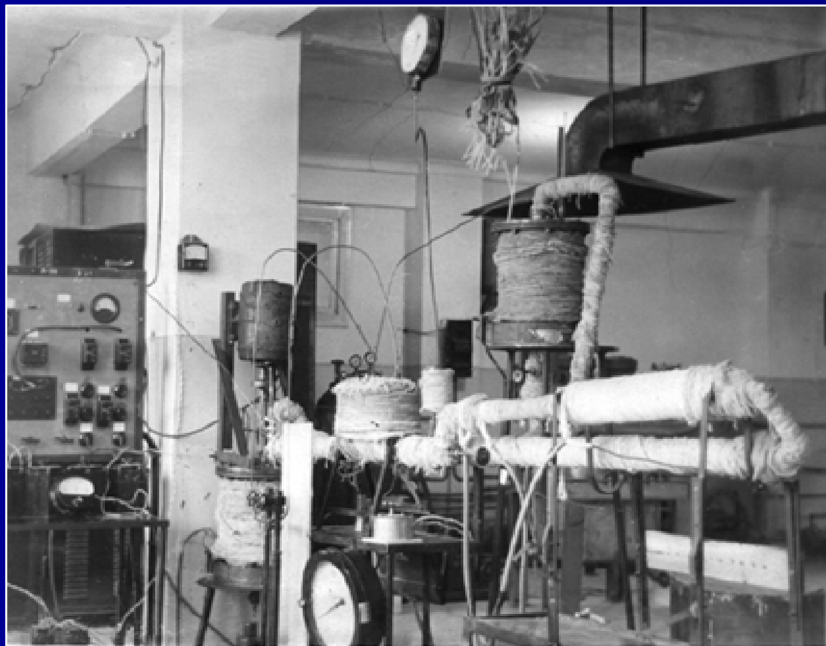
Russian Institutions	European Institutions
RRC Kurchatov Institute (Moscow) IPPE (Obninsk) EDB «Gidropress» (Podolsk) RIAR (Dimitrovgrad) NIKIET (Moscow) INR RAS (Troitsk)	ENEA (Italy)

PREPAREDNESS OF RUSSIA FOR THE PROJECT REALIZATION

Scientific and technological background of work

- more than 50-years experience with heavy metal coolant technology
- experience gained with development, construction, operation, and decommissioning of reactors for nuclear-powered submarines with lead-bismuth coolant and fast sodium reactors;
- designing and experience with development and construction of various types of power, research and experimental reactors;
- experience with design development of reactor systems with Pb-Bi coolant: SVBR-75/100; BRUS-150; BRUS-300, etc.
- experience gained with development, construction, and pre-operational adjustment/testing of target complex MC-1 for LANL (USA);
- databases (nuclear and reactor constants, material science, thermal physics, etc.)

50 YEARS HEAVY METAL COOLANT TECHNOLOGY IN INTENSIVE USE



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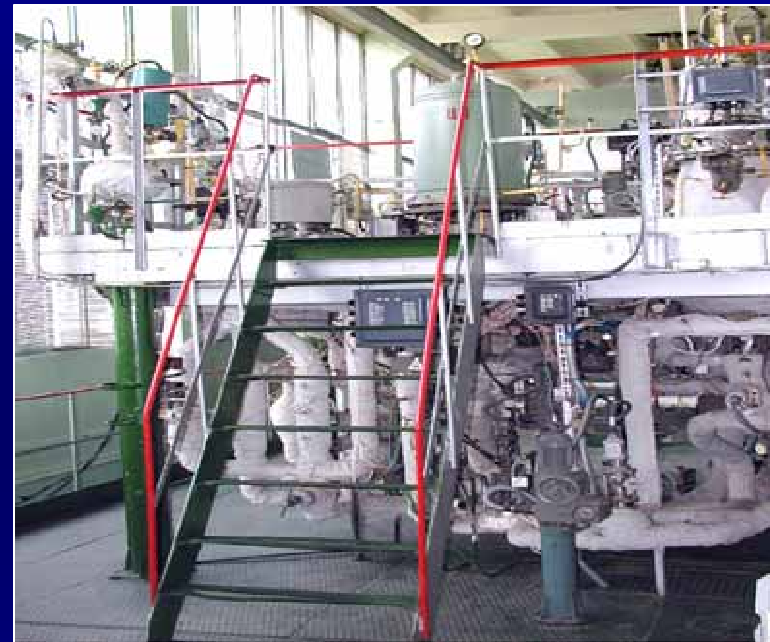
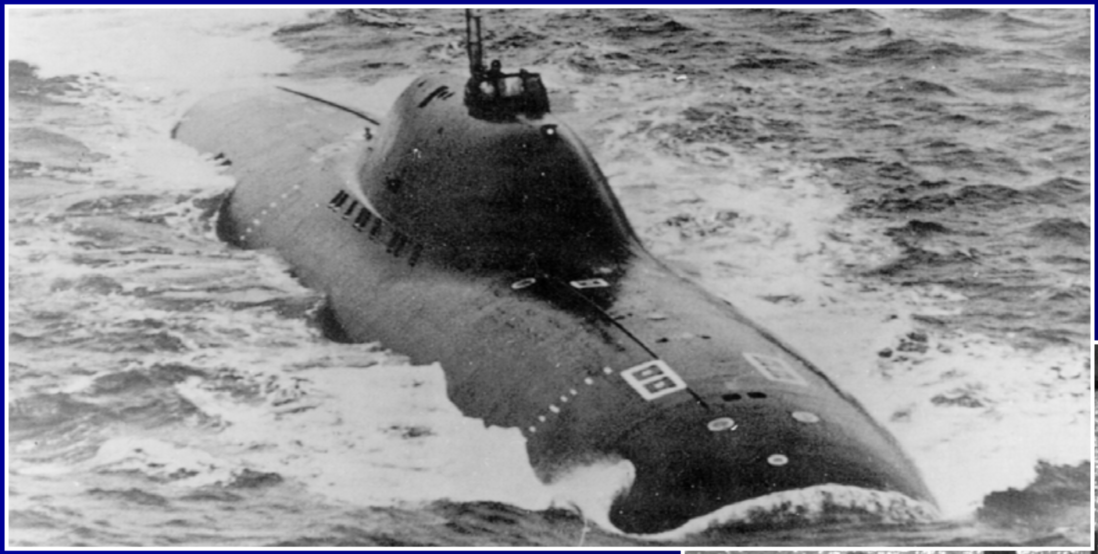
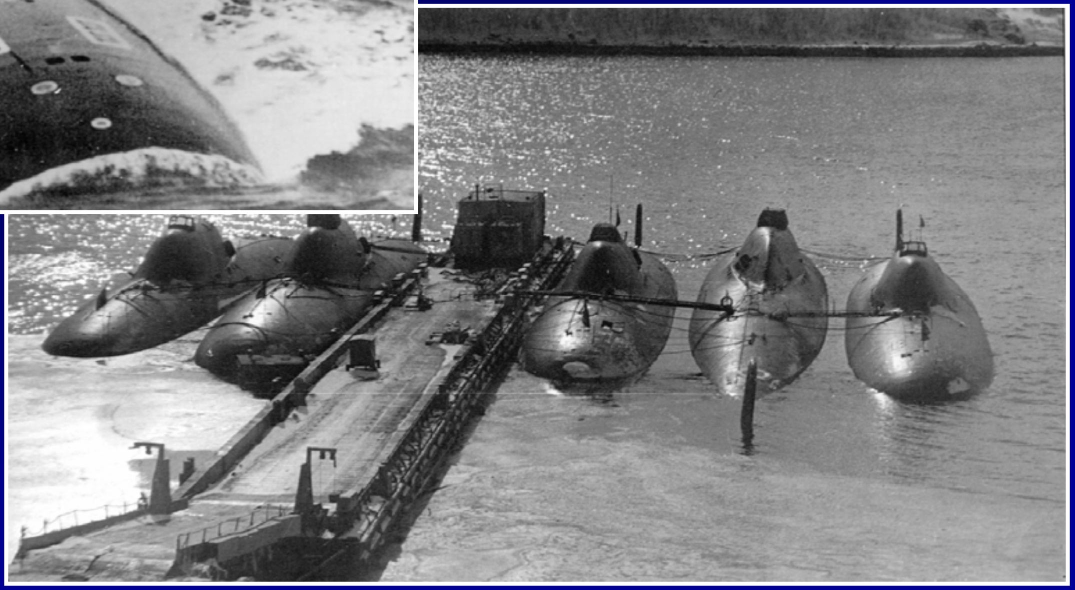


2001

Lead-bismuth facilities at IPPE

NUCLEAR-POWERED SUBMARINE
(LEAD-BISMUTH COOLED REACTOR)

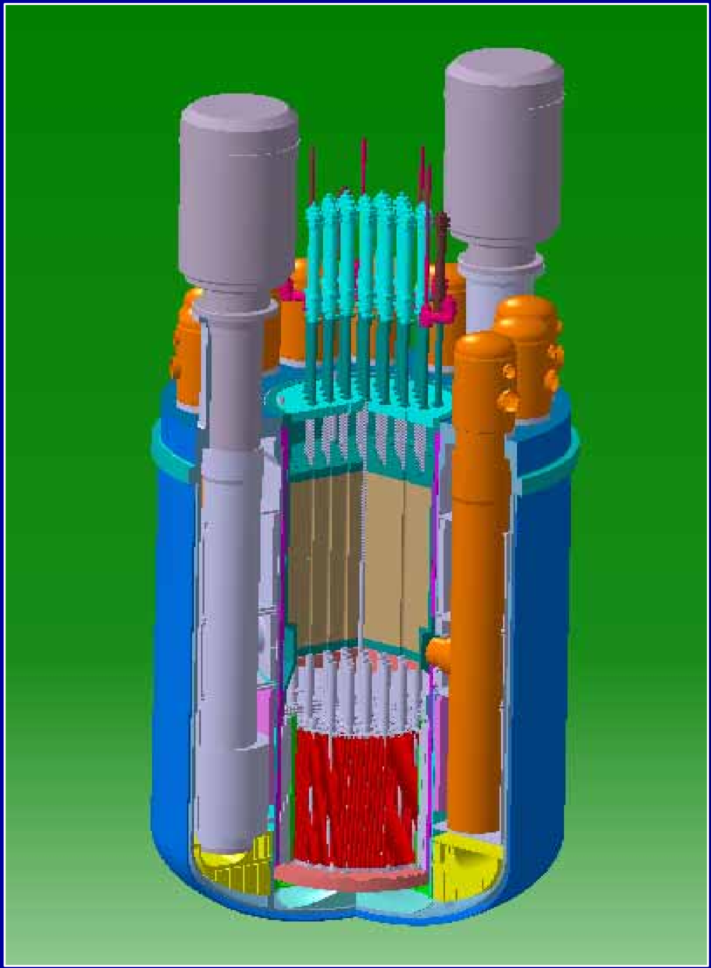




DEVELOPMENT OF CIVIL NUCLEAR POWER SYSTEMS WITH LEAD-BISMUTH COOLANT (DESIGNS)

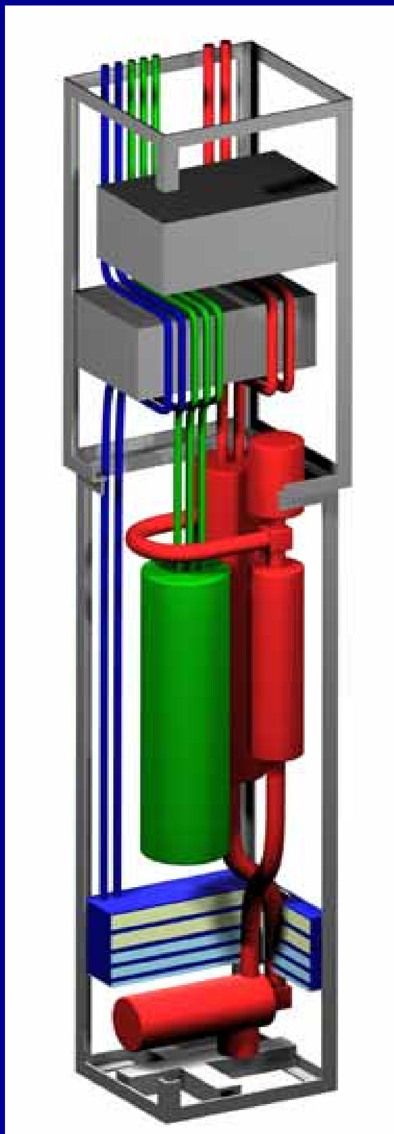
Reactor System	Thermal/Electric Power, MW	Years
TES-M	6/1	1987-1993
«Angstrem» (transportable NPP)	30/6	1985-1990
Cruise-50 (naval NPP)	200/50	1985-1990
SVBR-75	265/75	1995-2000
SVBR-75/100	260/100	2000-2003
TZMR-10	60/10	2002-2003
BRUS-150	550/150	1990-1995
BRUS-300	1100/300	1990-1995
SVBR-600	1875/600	1990-1995

SVBR-75/100 REACTOR



Parameter	Value
Thermal power (rated), MW	280
Temperature, °C at the core outlet at the core inlet	 482 320
Number of fuel rods, items	12114
Average volume power deposition density of the core, kW/dm^3	140
Average linear loading per fuel rod, kW/m	~24,3
Fuel type Loading of U-235, kg Average enrichment of fuel, at.%	UO_2 1470 16,1
Volume of coolant in primary circuit, m^3	18
Dimensions of reactor module: diameter by height, m	4,53×7,55

WORLD'S FIRST LEAD-BISMUTH TARGET COMPLEX OF 1 MW POWER FOR LANSCE PROTON ACCELERATOR



Designed jointly by IPPE and EDB "Gidropress"

Target complex at the final stage of assembling

Energy of protons	800 MeV
Current	1.25 MA
Power of proton beam	1 MW
Maximum density of current	77 mkA/cm2
Yield of generation neutrons	$1.5 \cdot 10^{17}$ N/s
Target inlet temperature	230 ^{0}C
Target outlet temperature	330 ^{0}C
Coolant Flow rate	14.2 m^3/h

NWB PROJECT EXPERIMENTAL BASE IN RUSSIA

- Reactor test complex involving: the high-flux research reactor SM-3; fast neutron reactor BOR-60 with sodium coolant, etc.;
- Reactor research complex involving: super-power pulsed reactor BARS-6 and zero power reactors;
- Technological research complex involving: experimental rigs for bringing to wide use the large-scale equipment for lead-bismuth coolant systems;
- Radiochemical laboratory complex for complete analysis of irradiated materials and irradiated fuel of nuclear reactors;
- Research material irradiation complex for a full post-reactor studies irradiated fuel elements, SA, and reactor materials;
- Experimental semi-industrial complex for production and reprocessing of fuel compositions (including compositions with minor actinides).

Institute for Physics and Power Engineering
(IPPE, Obninsk)

- BFS Zero power facility
- Pulsed reactor BARS-6
- Radiochemical complex
- Lead-bismuth technological test facilities
- Accelerator EGP-15

Research Institute of Atomic Reactors
(RIAR, Dimitrovgrad)

BOR-60 reactor

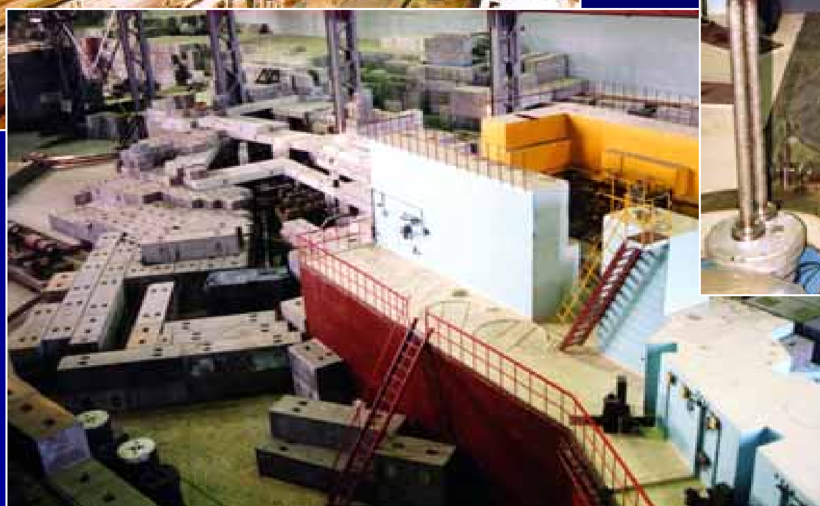
Chemical/Technology facilities

BN-600 fuel assembly fabrication line

Hot cells

Moscow Mezon Plant (INR RAS, Troitsk)

High-current proton accelerator
Experimental hall with beam splitting
Neutron sources and targets of proton accelerators

PREPAREDNESS OF EUROPEAN LABS

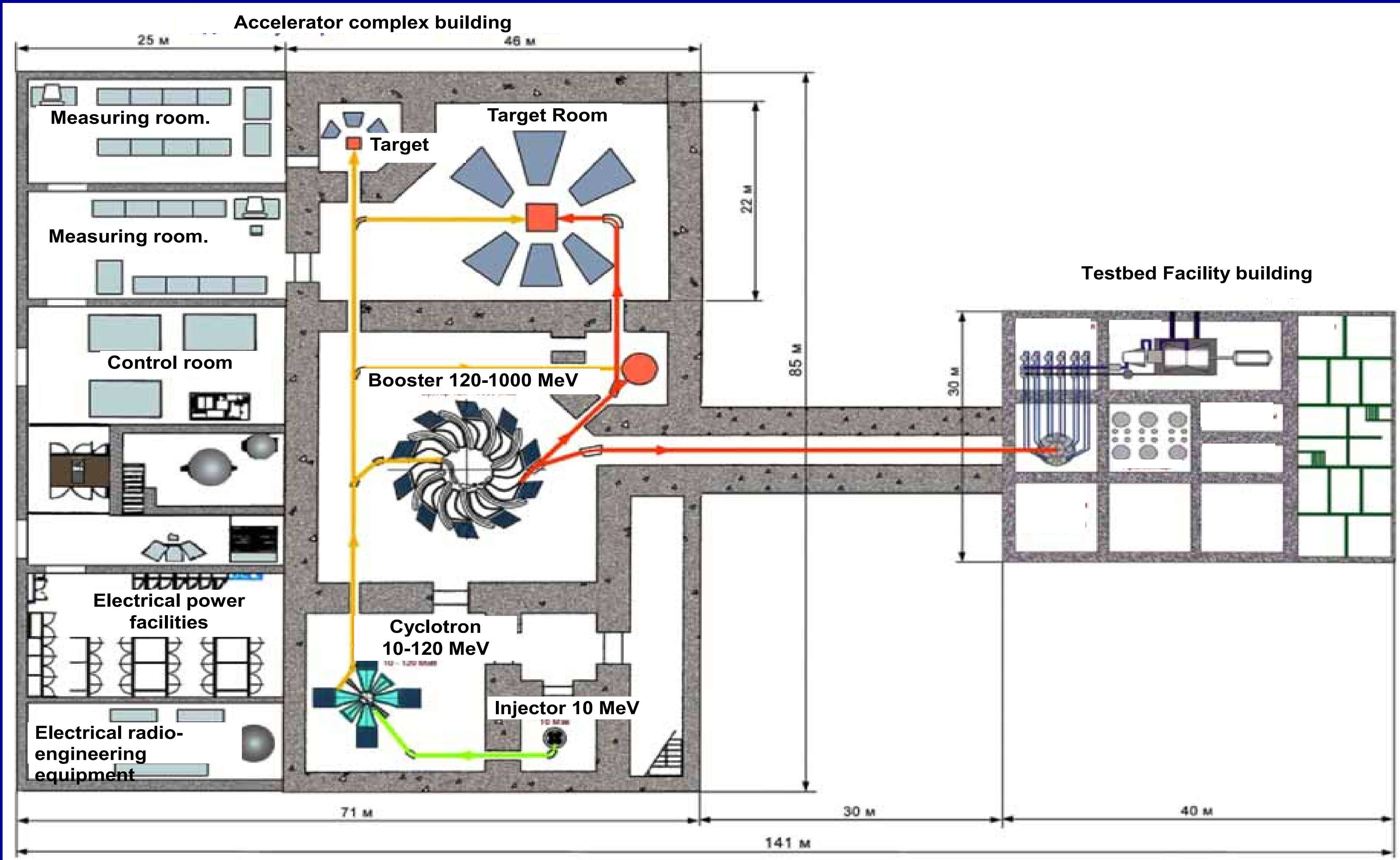
- ENEA/ANSALDO led design of 1-3 MW power proton cyclotron

- On-going projects
  - EAP-80, XADS; MYRRHA; MEGAPIE ; TRADE etc.

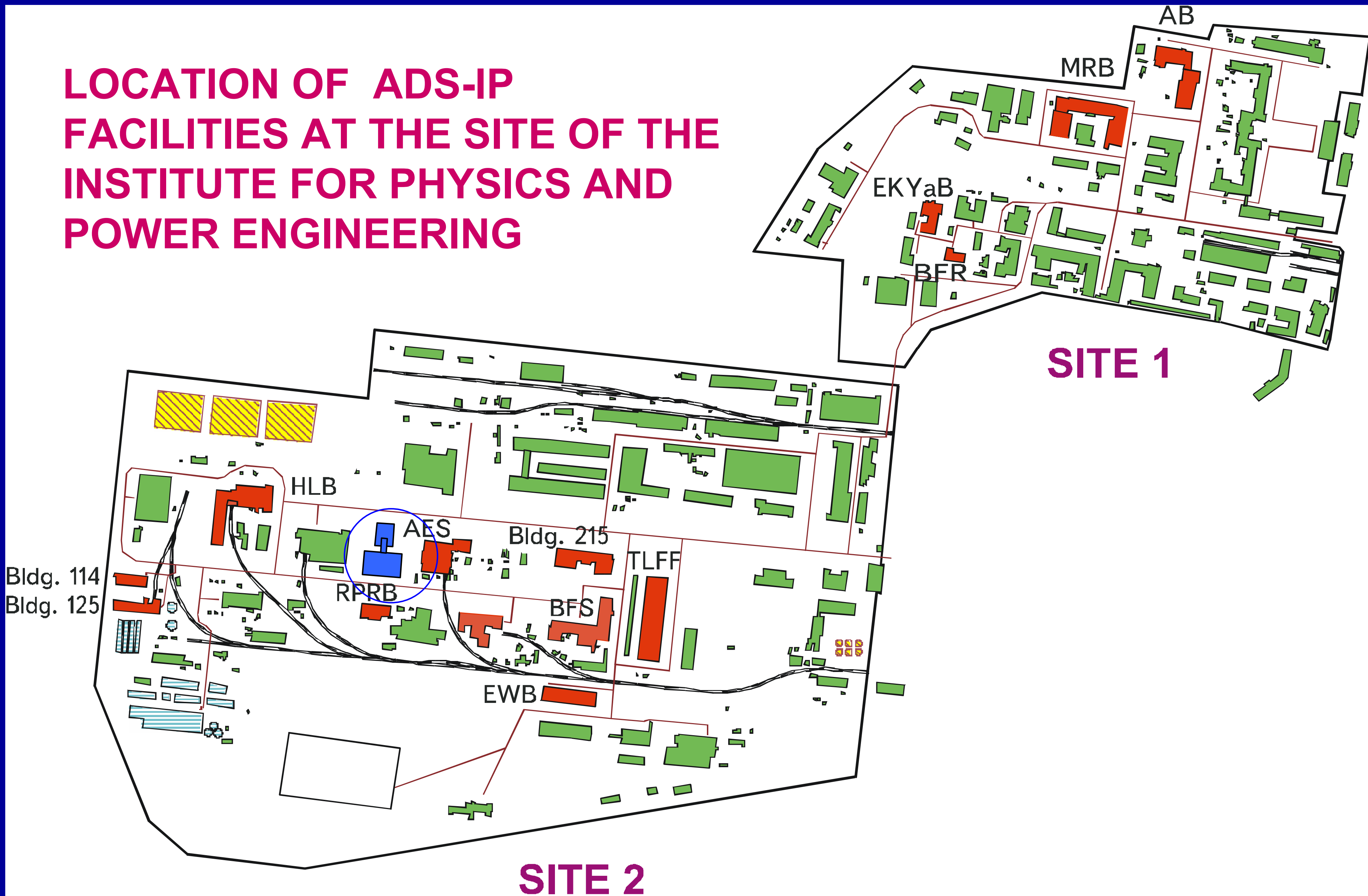
- R&D in the frame of "European RoadMap for Accelerator Driven Systems (ADS) for Nuclear Waste Incineration"

- Large-scale experimental facilities on coolant technology
  - KALLA lab in Karlsruhe, Germany
  - CIRCE in Brasimone, Italy

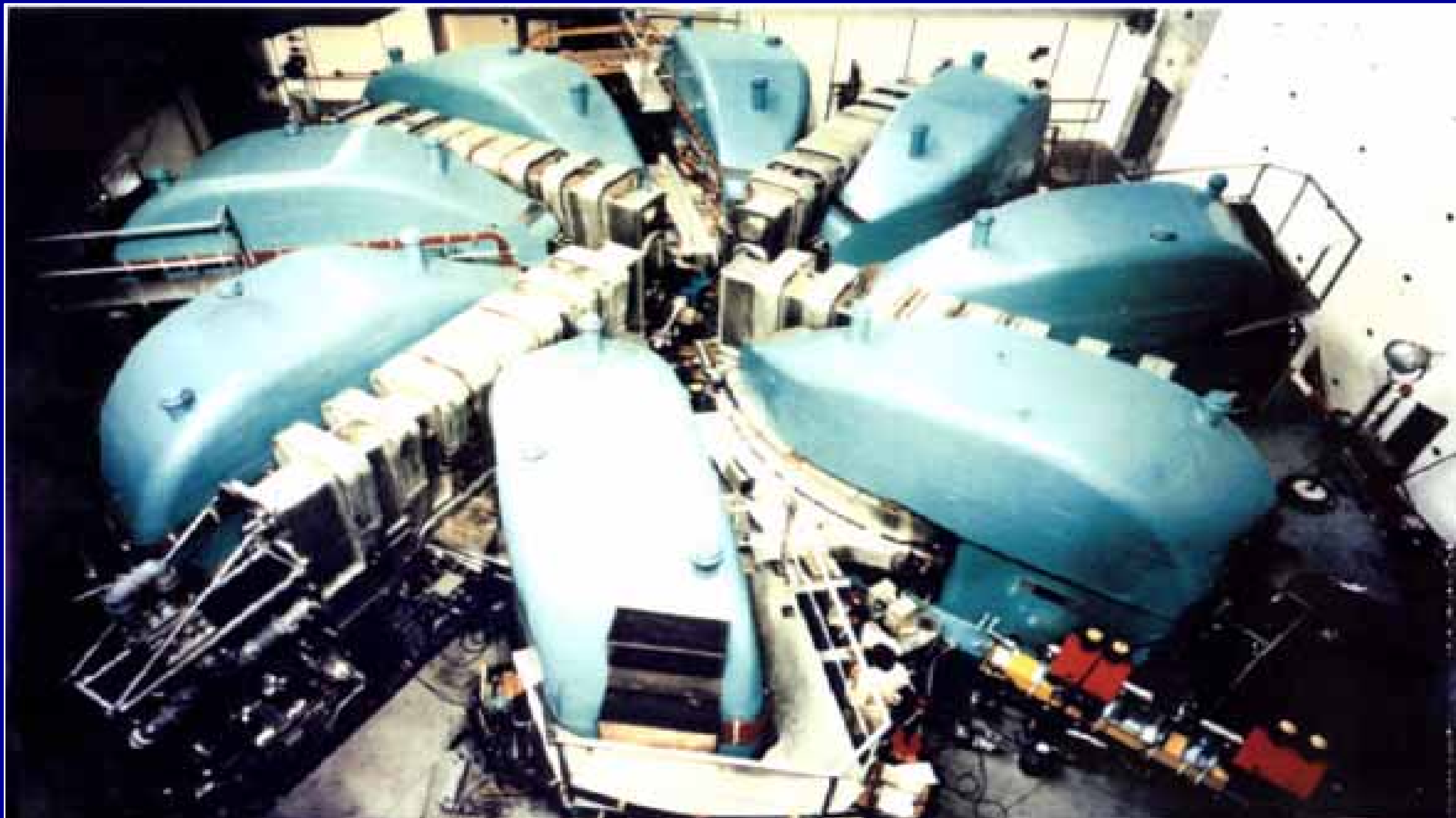
LAYOUT OF NWB INDUSTRIAL PROTOTYPE



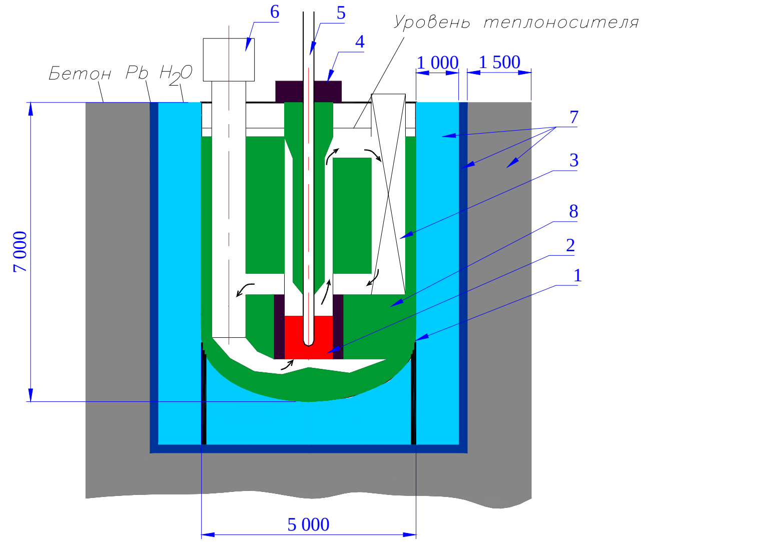
LOCATION OF ADS-IP FACILITIES AT THE SITE OF THE INSTITUTE FOR PHYSICS AND POWER ENGINEERING



A 600 MeV cyclotron: PSI as a model?

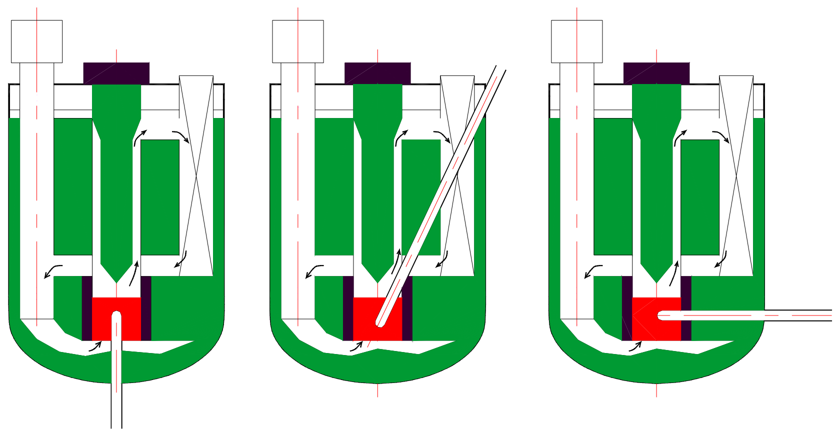


Maximum beam power ~ 1 MW



Мишенно – бланкетный модуль

1– Корпус моноблока 2– Активная зона 3– Теплообменник
4– Перегрузочное устройство 5– Протоновод 6– Циркуляционный насос
7– Внешняя защита 8– Внутрикорпусная защита

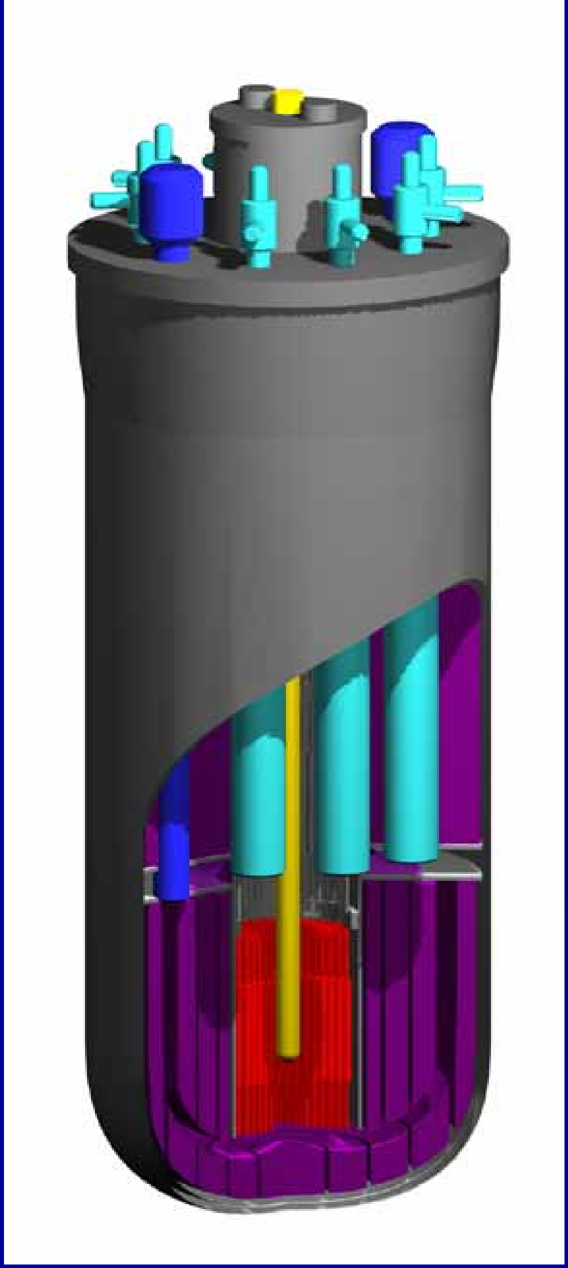


Варианты ввода пучка протонов

SHEME OF THE SUBCRITICAL REACTOR AND TARGET

Maximum use of the experience of the facilities with lead-bismuth coolant, including the SVBR 75/100 reactor

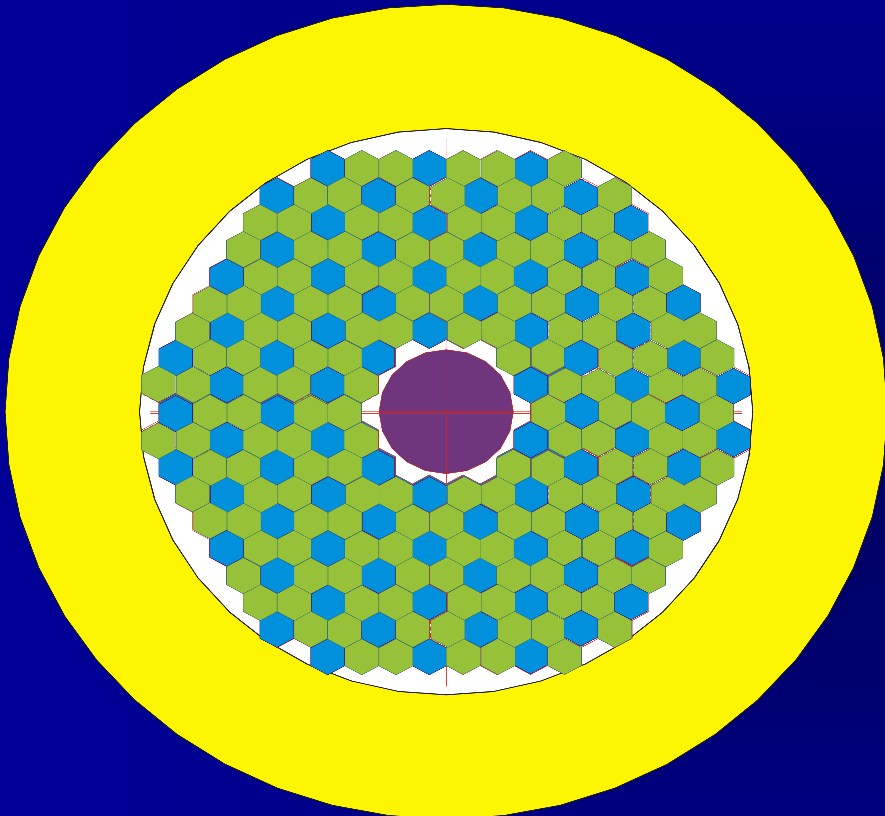
- **Mono-block construction of NWB comprising the whole primary circuit with its heat-exchange equipment, in-vessel radiation shielding, subcritical reactor and the target modules**
- **The reloading device is mounted on the cover of the module for loading/unloading of the subassemblies**
- **Temporary storage of subassemblies with SNF**

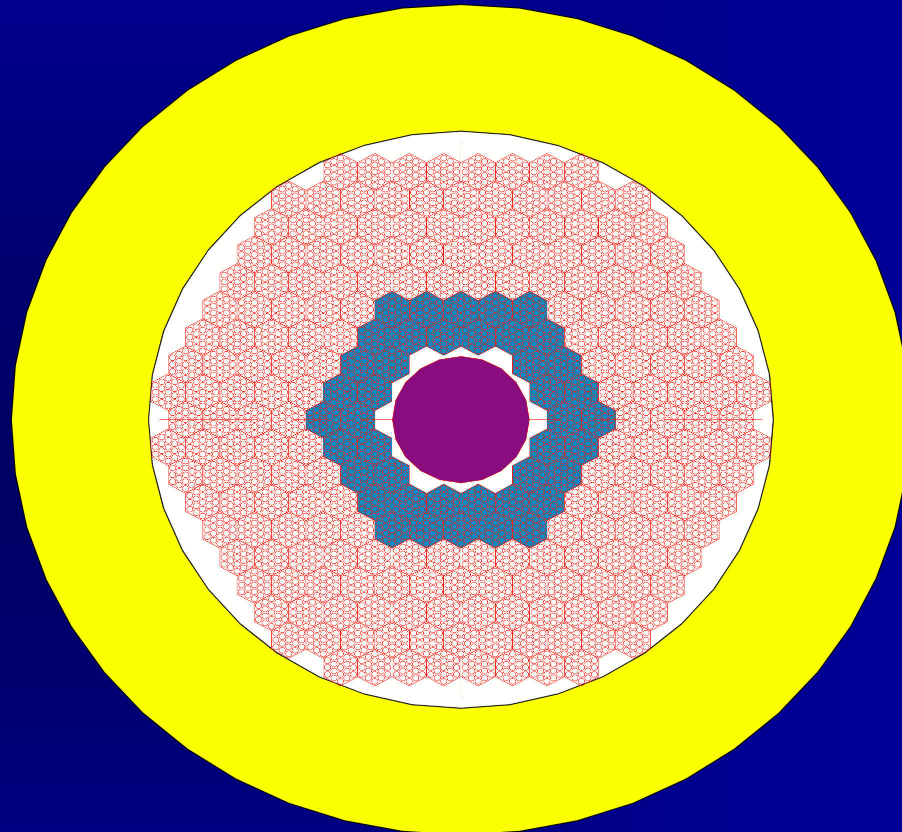


NUCLEAR WASTE BURNER PARAMETERS

Thermal power of target	3 MW
Thermal power of reactor	100 MW
Dimensions of reactor module: diameter x height, mm	985x900
Coolant	Pb-Bi
Temperature	
core inlet	270°C
core outlet	435°C
Average velocity of coolant	2 m/s
Fuel -	UO_2/ UN / U-Zr
Loading of U (h.a.)	3000 kg

NWB NEUTRONICS





#1 version

MA Dioxide + ZrO_2

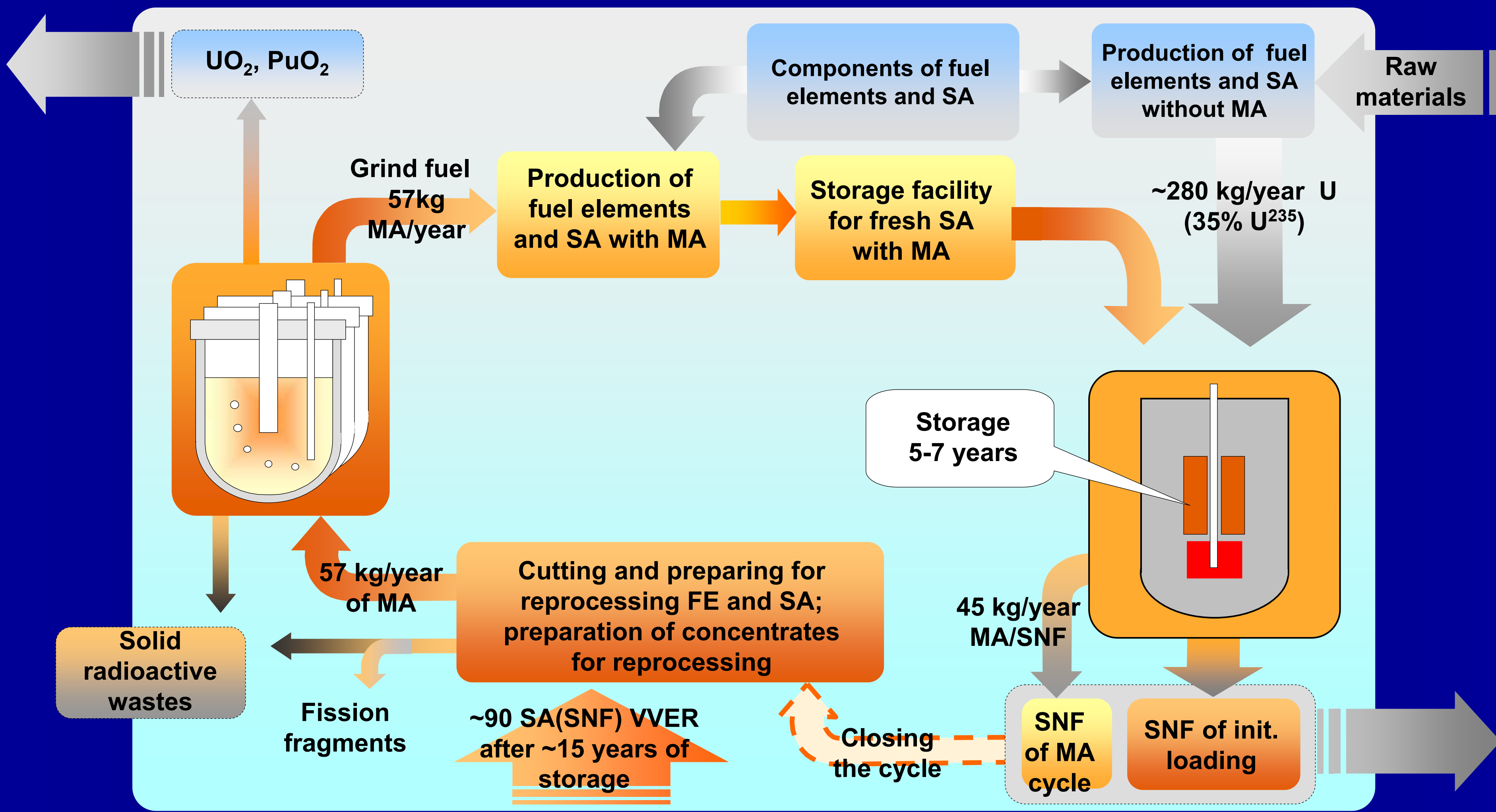
#2 version

AmO_2+NpO_2+Zr

NWB NEUTRONICS (cont.)

Parameter, units	Variant 1		Variant 2	
Fuel with MA (400 kg)	MA Dioxide + ZrO_2 (108 assemblies)		AmO_2+NpO_2+Zr (42 assemblies)	
Loading of U (h.a.), kg	2375		2924	
Enrichment of ^{235}U, %	22, 25.5, 38		18.5, 21, 25.5	
Beginning/end of service time between loadings (year)	T = 0	T = 1	T = 0	T = 1
Multiplication factor M	29,643	24,552	29,39	30,143
K_s	0,967	0,961	0,967	0,968
Burning of MA, kg/year	15,1		15,5	
Specific burning of MA, g/MW-day	~ 0,41		~ 0,43	
Buildup of ^{238}Pu kg/year	10,1		9,9	

NWB FUEL CYCLE



THE MAIN PROBLEMS AND KEY SOLUTIONS

Key engineering, technical and technological solutions	Solution versions			
Coolant type in the reactor and target cooling circuit	Pb-Bi			
Primary circuit configuration	Integral		Loop type	
	In the case when the problems of radiation shielding and equipment maintenance on the reactor - target circuit cover require "extremely" heavy shielding, it is reasonable to consider the loop-type configuration			
Target design	"Window"		"Windowless"	
Target cooling circuit	Independent		Combined with the reactor cooling circuit	
Number of heat – exchanging circuits	2		3	
Primary pump type	Rotary (centrifugal) pump with electric drive			
Fuel type	Vibro - MOX	U-Zr	UN	UO_2
	The main part of the reactor is loaded with a "manoeuvrable" fuel (UN, U-Zr or UO_2, vibro - packed). As the operational and engineering problems are being solved a partial loading of assemblies with MA may take place. Isotopic composition of SA with MA has to be determined based on the reprocessing technology			
Core design and its refueling	Assembly – by – assembly refueling		Full core loading, assembly – by – assembly unloading	

THE MAIN PROBLEMS AND KEY SOLUTIONS (cont.)

Fuel assembly design	Fuel pin spacing – rib - to - rib	
Fuel pin design	Cladding consists in a four – rib tube, gas – bonded or sodium – bonded	
Safety rods and shim rods	present	absent
	The solution about the design with or without safety and shim rods depends significantly on the accuracy of calculation justification of reactor subcriticality level	
In – vessel shielding	B_4C bricks cooled with the blanket	
Equipment shielding in the area of neutron guide tube	The necessity to have additional shielding against the secondary coolant activation and to provide the acceptable radiation environment on the cover of the reactor – target module in order to have the access to the equipment requires certain lay-out solutions different from the ones accepted in the reactor building industry	
Increased gas – system activity in the target circuit	The increased gas system activity requires the development of new methods to control leak – tightness of fuel pin cladding, e.g. spectrometric radiometer of high accuracy	
Proton beam interruption problem	A relatively high frequency of proton beam interruption requires special measures to decrease thermal – cycling loads on the primary circlet equipment. The use of fuel with high heat conductivity is considered of a high priority	
Coolant technology and materials	The effect of spallation products on the material corrosion resistance, the effect on coolant quality control, including the devices and techniques for cleaning the circuit, coolant and gas system from impurities require additional study	

FUEL AND CLADDING MATERIALS

- corrosion resistance of cladding at operation temperatures
- radiation stability and strength at neutron high doses
- multiple and thorough thermal power change caused by beam trips
- high radiation stability of fuel

Cladding	Four-ribs tube of EP-823
Fuel	Vibro-packed MOX or Cermet
Gap	Na for Cermet

PRIMARY CIRCUIT STRUCTURE MATERIALS

Component	Material	Design basis limits
Cladding	EP-823	Corrosion resistance validated on the basis of 5 000 hours at 650 ^{0}C; 20 000 hours at 560-600 ^{0}C Data base on steel properties at the affecting doses up to 100 dpa at the temperature of 300-600^o C
In-vessel structures	EP-302 (new developments on 9% Cr steels)	Corrosion resistance validated on the basis of 100 000 – 120 000 hours at the temperatures of 450-500 ^{0}C
Steam-generator tubes	bimetal EP-302/CHS-33 (new developments on 9% Cr steels)	
Primary vessel	X18H10T Stainless steel	Durable corrosion resistance validated up to the temperatures of 400-420 ^{0}C The legitimate fluence of fast neutrons of up to $\sim 5 \cdot 10^{23}$ n/(cm^{2}s) was validated by the experience of BR-10 reactor operation at IPPE

POWER RELEASE IN MA DIOXIDES OF DIFFERENT CONTENT

(theoretical density γ_{th} = 11,46 g/cm^3)

NUCLIDE	MA = Am		MA = VVER SNF		MA = "Spiro"mixture	
	weight %	q, W/cm^3	weight %	q, W/cm^3	weight %	q, W/cm^3
^{237}Np	-	-	43.42	8.975E-05	12.5	2.58E-05
^{241}Am	87	1.021	48.1	0.565	50	0.587
^{242m}Am	0.1	-	0.044	2.077E-05	-	-
^{243}Am	12.9	0.0085	7.14	0.0047	25	0.0165
^{242}Cm	-	-	0.0001	1.375E-03	-	-
^{243}Cm	-	-	0.0205	3.889E-03	-	-
^{244}Cm	-	-	1.21	0.354	12.5	3.657
^{245}Cm	-	-	0.08	4.734E-05	-	-
Total power release at γ_{th}, W/cm^3		1.03		0.93		4.26

ISOTOPIC CONTENT OF FUEL in SA

FUEL ⇒ ⇓ NUCLIDE (kg)	SA with MA, variant 1		SA with MA, variant 2		SA with MA, variant 3		SA with UO_2, variant 1	
	loading	unloading	loading	unloading	loading	unloading	loading	unloading
^{234}U	-	0.019	-	0.22	-	0.042	-	-
^{235}U	-	-	-		-	-	4.96	3.19
^{236}U	-	-	-		-	-	-	0.32
^{238}U	-	-	-		-	-	9.34	8.75
^{238}Pu	-	0.71	-	0.80	-	1.47	-	0.0008
^{239}Pu	-	0.038	-		-	0.061	-	0.37
^{240}Pu	-	0.016	-		-	0.008	-	0.014
^{241}Pu	-	0.001	-		-	1.9E-04	-	3.6E-04
^{242}Pu	-	-	-		-	3.5E-06	-	6.4E-06
^{237}Np	1.59	1.14	1.59	0.977	6.15	4.433	-	0.0108
^{241}Am	1.79	1.21	1.79	0.987	2.93	1.983	-	2.9E-05
^{243}Am	0.27	0.21	0.27	0.19	0.44	0.355	-	1.7E-07
^{242}Cm	4.1E-06	0.041	4.1E-06	0.048	-	0.056	-	4.2E-07
^{243}Cm	0.001	0.004	0.001	0.005	-	0.003	-	7.8E-08
^{244}Cm	0.046	0.071	0.046	0.072	-	0.045	-	1.0E-07
^{245}Cm	0.003	0.006	0.003	0.007	-	0.002	-	8.9E-08
Fractions	-	0.23	-	~ 0.3	-	1.055	-	1.66
Total number of MA	3.70	2.68	3.70	2.3	9.52	6.88	-	0.0108
MA burning, kg/SA(%)	-	1.02 (27.6)	-	1.4 (37.8)	-	2.64 (27.2)	-	-
Uranium	-	0.019	-	0.022	-	0.034	14.3	12.25
Plutonium	-	0.765	-	~ 0.85	-	1.25	-	0.38

CURRENT RUSSIAN TECHNOLOGIES: MA SEPARATION AND FUEL FABRICATION

In Russia only ^{237}Np (in the form of NpO_2) and, periodically, americium-241 minor actinides are separated when fuel is regenerated.

Variants of composition of the fuel with MA:

- Fuel with the artificially made mixture of Np and Am.
- Fuel with neptunium only.
- Fuel with americium only.
- Fuel with a mixture of MA from SNF.

In all cases, MA will most likely be in mixture with matrix material in the form of oxides, nitrides or metals. Initial content of MA in the fuel should be determined with the account of the limit on the specific heat generation in it of ~0.2 W/cm^3.

CURRENT RUSSIAN TECHNOLOGIES:
FABRICATION OF FUEL SUBASSEMBLIES WITH MA

At this time, the amount of separated actinides is not large (tens of kg) and there is no actual fabrication of fuel elements and subassemblies with MA.

MA burnup process in an ADS system should have several steps:

- loading of one or several experimental assemblies into the core;
- increase of the number of assemblies with the considerable increase of MA amounts in the core.

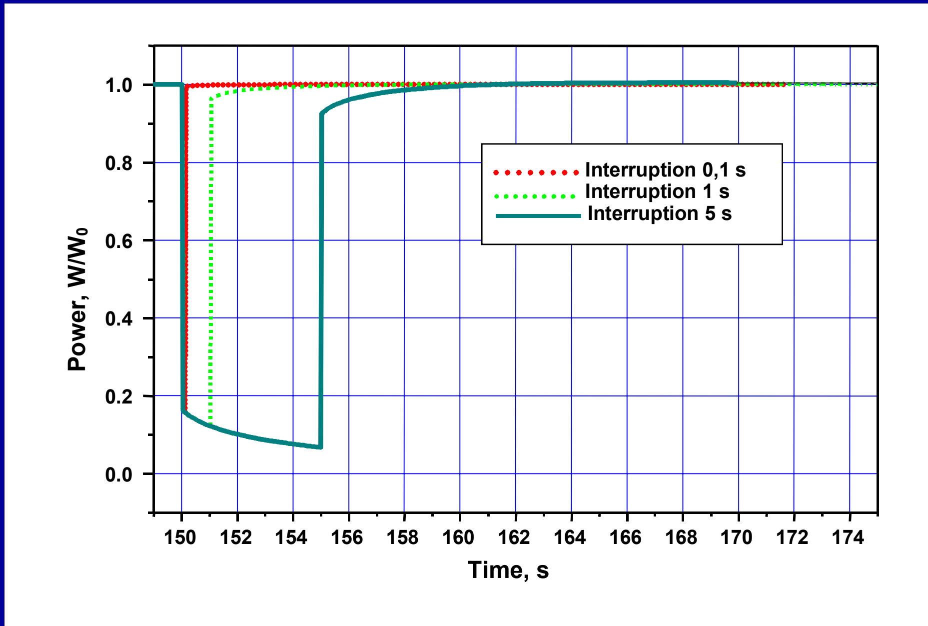
Individual assemblies can be fabricated at RIAR and partially at IPPE using vibropack technology.

CURRENT RUSSIAN TECHNOLOGIES: REPROCESSING IRRADIATED ASSEMBLIES

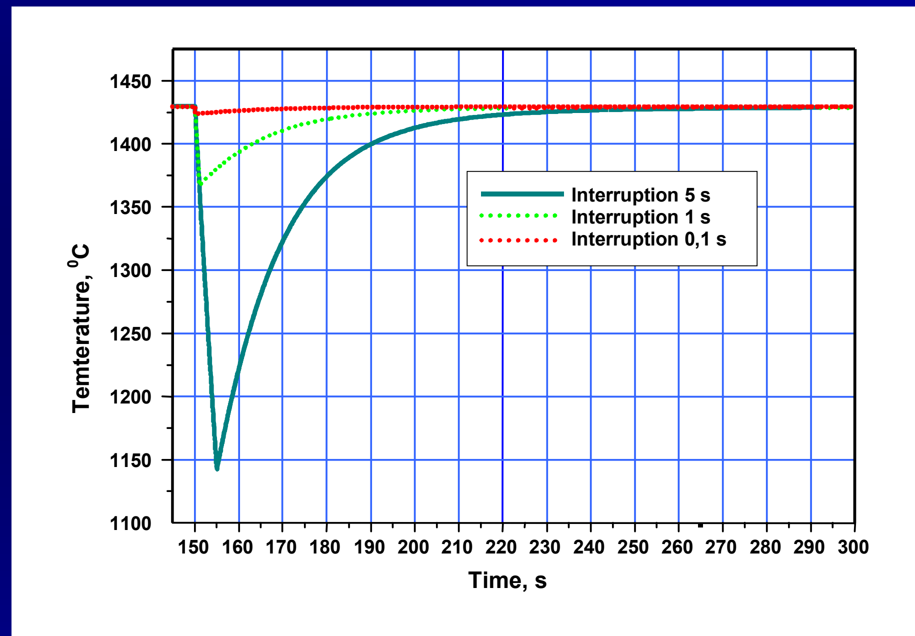
On an industrial scale, this procedure should be carried out at the specialized enterprises of Russia. However, there is no industrial technology for reprocessing of irradiated assemblies so far.

Individual fuel elements and SA may be reprocessed at RIAR and IPPE. This activity will provide experimental assessment of calculated data on transmutation and allow to make recommendations for industrial enterprises.

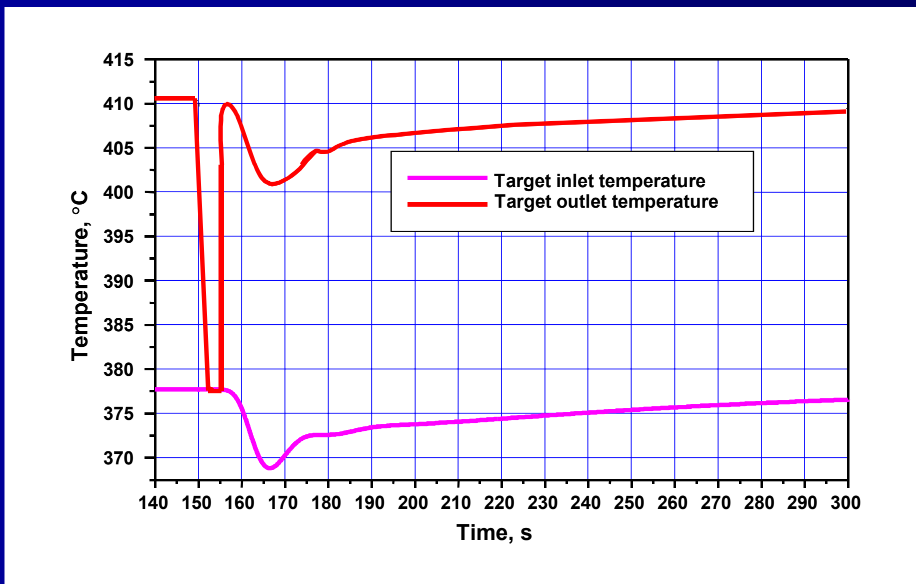
PROTON BEAM INTERRUPTIONS



Power of the reactor



Average fuel temperature

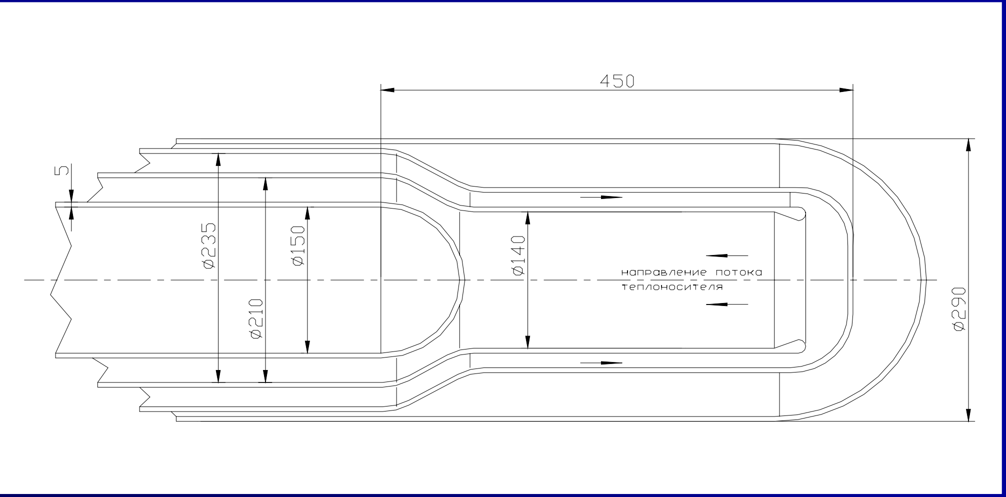


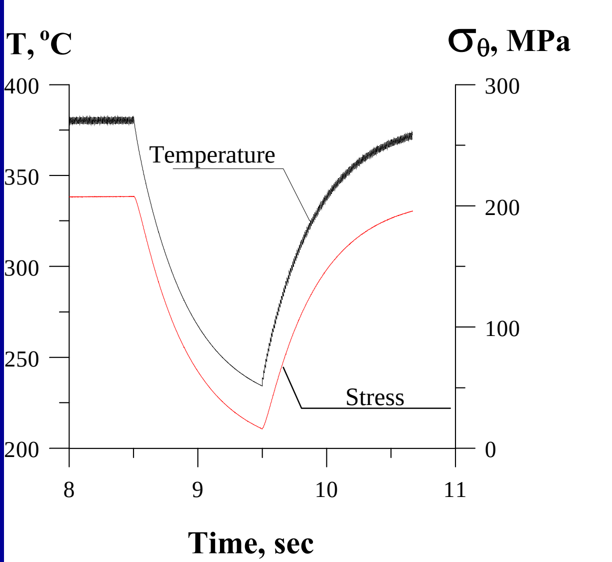
Inlet and outlet temperatures of the target (interruption - 5 s)

THERMAL MECHANICS, STRENGTH AND RELIABILITY OF THE TARGET AND PRIMARY CIRCUIT EQUIPMENT

Acting factors

- radiation damage of materials,
- corrosion damage in Pb-Bi,
- high energy release in the window and coolant
- coolant pressure
- pulse nature of proton beam and related stress and temperature loads





Parameters of target window

Energy release, kW	25
Material	EP-823
Radiation dose, dpa/year	70
Build-up, appm	
He	2000
H_2	20000
Coolant velocity, m/s	2
Window temperature, ^0C	400-500
Life time, years	1

Pb-Bi COOLANT AND STRUCTURAL MATERIALS: CORROSION AND MASS TRANSFER

Peculiarities of Pb-Bi	Consequences
A lot of chemical elements and compounds are well soluble in Pb-Bi	Destruction of materials and tightness failure
Liquid Pb-Bi evidently interacts with oxygen and its concentration in coolant is the most important factor!	Excess of oxygen results in slagging and the coolant thermal hydraulics degradation. Lack of oxygen led to dissociation of oxide films and progress of corrosion processes

RADIATION SAFETY

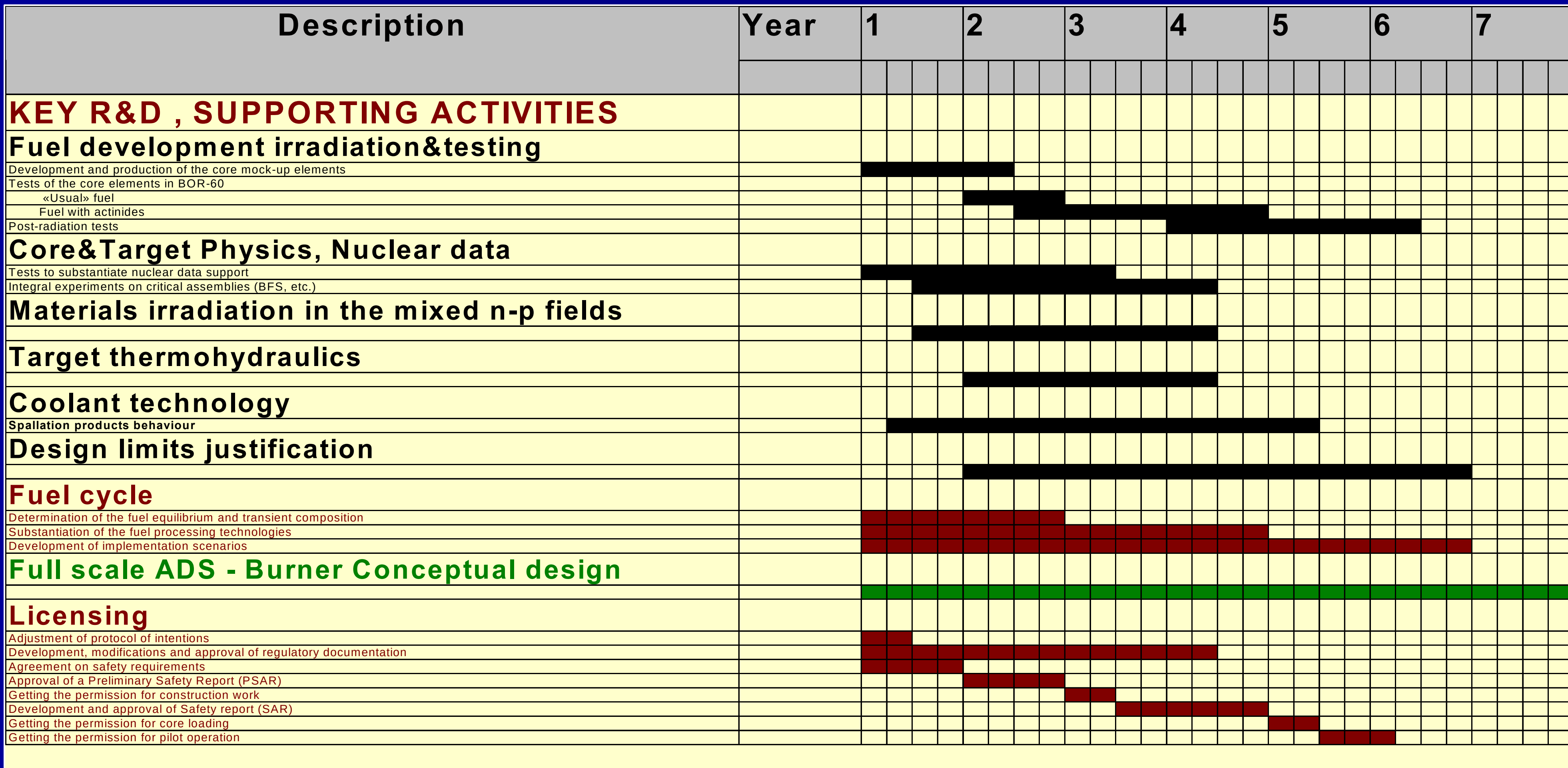
<table>
<tr><td colspan="2">Gamma-radiation dose rate on the cover of the core with the vertical insertion of a beam after 3-year operation and 30-day holding is ~1R/hour</td><td colspan="2">The other ways of beam insertion should also be considered</td></tr>
<tr><td colspan="2">External radiation protection of the target vessel:
Water tank ~1000 mm
Pb layer ~200 mm
Heavy concrete ~1500 mm</td><td colspan="2">▪ It will be possible to reduce radiation dose rate to 0.3 μR/s</td></tr>
<tr><td colspan="2" rowspan="5">Accumulation and release of spallation products into the gas system</td><td>Element</td><td>Release into gas system</td></tr>
<tr><td>Polonium</td><td>$5 \cdot 10^{-9}$ %</td></tr>
<tr><td>Mercury</td><td>0.1 %</td></tr>
<tr><td>Boron and iodine</td><td>$6 \cdot 10^{-3}$ %</td></tr>
<tr><td>Rubidium and cesium</td><td>10^{-3} %</td></tr>
</table>

SUPPORTING EXPERIMENTS AT THE ACCELERATOR OF MOSCOW MESON PLANT

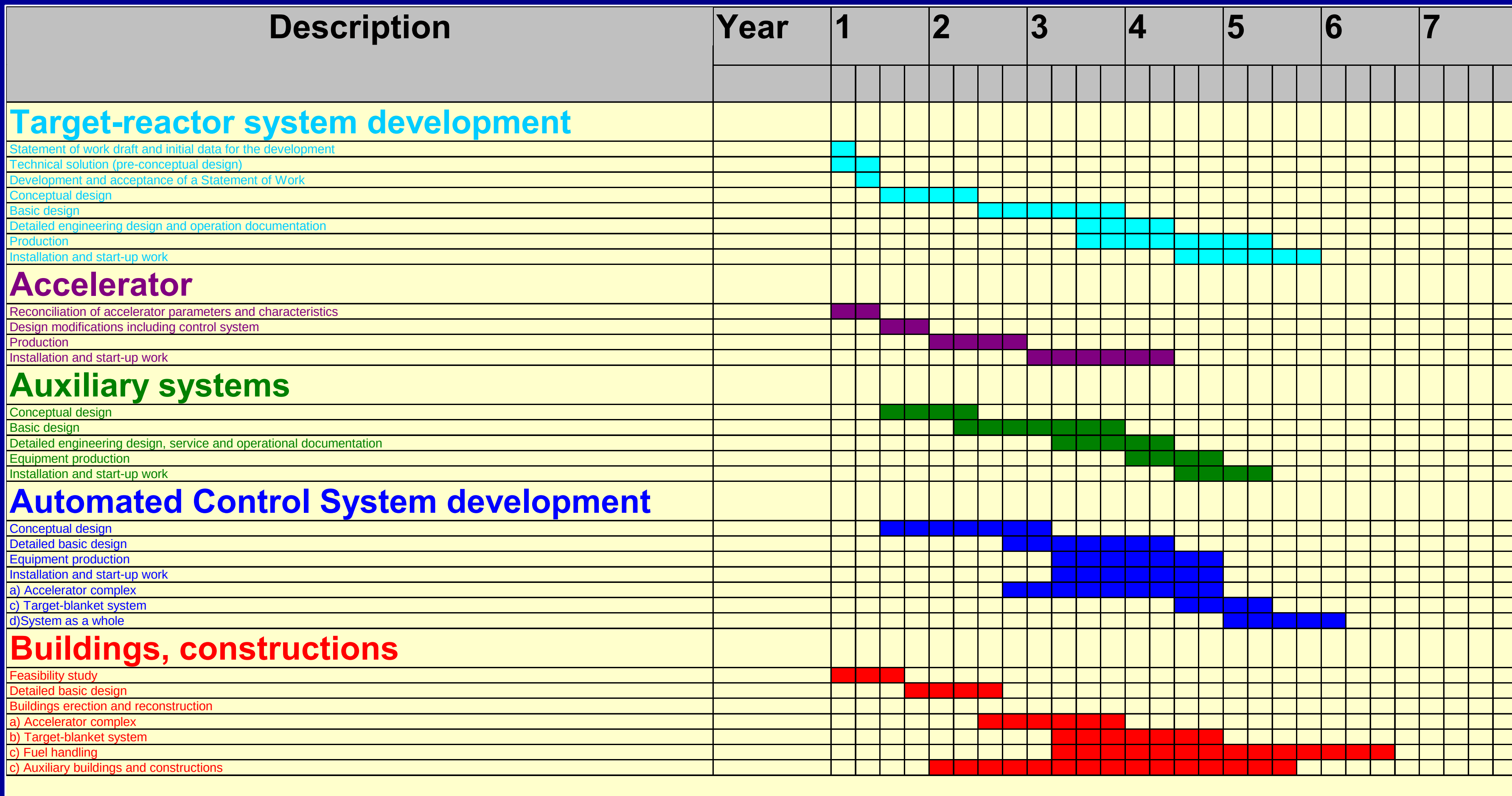
View of neutron source complex

- Neutronic studies using an enhanced spectrometer.
- Designing an experimental target with Pb-Bi coolant with the power of ~30KW.
- Studies of damages of “window” structural materials in a proton beam.
- Studies of radiation safety.

NWB PROJECT: PRELIMINARY SCHEDULE



NWB PROJECT: PRELIMINARY SCHEDULE (cont.)



CONCLUSIONS

- Proposal for launching an international project for the creation in Russia of the world's first Accelerator - Driven System and semi-industrial demonstration of the technology for safe transmutation of MA and other high-level radwastes have been formulated.

- The key engineering, technical, and technological challenges to be solved, preliminary parameters of the industrial prototype, scope of R&D works, as well as the capacity of the Russian labs having long-term experience in designing, manufacturing, and operating the facilities has been outlined.

- It has been shown that 100 MW Nuclear Waste Burner can be created within 7-8 years; with the loading of 200 - 400 kg of MA it could provide the burning level of ~ 10 - 20 kg/year.

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