The Abdus Salam
International Centre for Theoretical Physics

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Joint ICTP-IAEA School of Nuclear Energy Management

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Innovative Nuclear Systems

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Innovative Nuclear Systems
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Nuclear Power Technology Development Section

Division of Nuclear Power

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International Atomic Energy Agency

Nuclear innovation is hard

- Very conservative enterprise
- Cost of designing and certifying a new reactor concept is approximately $1B
  - Standardization vs. innovation
- Advanced nuclear technology is not the only need
  - Communication tools
  - Regulatory tools for adapting to innovation
  - Construction techniques
  - Lessons from construction and operational experience
  - Experimental and analytical tools
- How can a Gen-IV or other advanced reactor economically compete against Gen-III or III+?

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Trends in nuclear innovation

- Technology transfer from another field
  - Exascale computing
  - Nanomaterials
  - Instrumentation and control
  - Power generation
- "Frankenstein" approach—combining the best parts to produce something powerful, some refinement needed
  - VHTR+MSR-He-UF4=AHTR
  - VHTR+SFR-Na=GFR
- Technology focused on policy goals
  - Safety
  - Nonproliferation
- Cost-effective modernization of existing plants

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To paraphrase Admiral Rickover:

An academic reactor is simple, small, cheap, light, flexible of purpose, requires little development, and can be built quickly. A practical reactor is large, heavy, complicated, expensive, is being built now, is behind schedule, and is requiring an immense amount of engineering development on seemingly trivial items.

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GIF (Generation IV International Forum) Strategy

Identify challenging goals

Create a multilateral framework for cooperation among leading nuclear development nations

Organize, grow, expand membership

<u>Goals</u>

Sustainability: resources and waste management

Economics: Competitive cost and financial risk

Safety: Reliable, low risk of core damage, no EPZ

Proliferation: Not a weapons path; secure against terrorism

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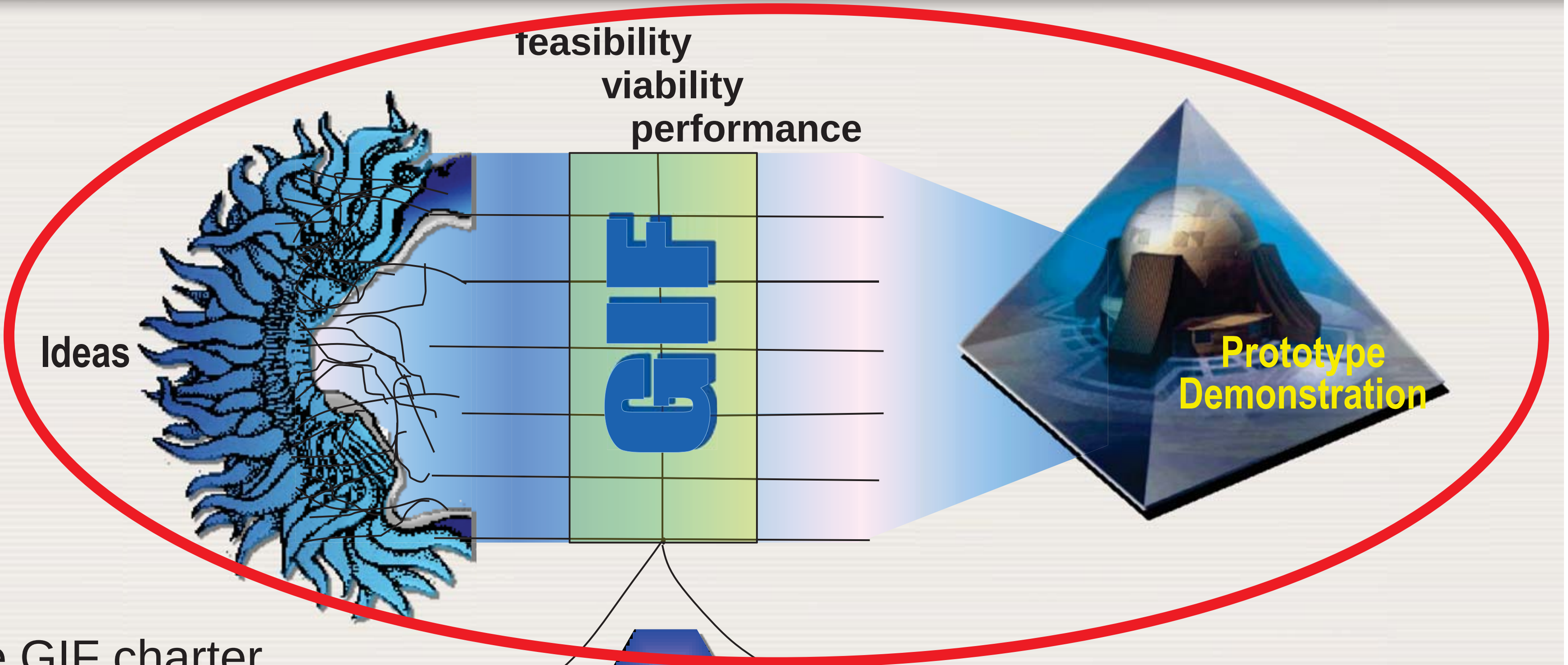
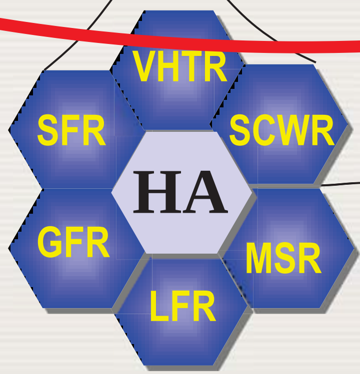
From concept to demonstration



The GIF charter encompasses the complete development cycle and fuel cycle

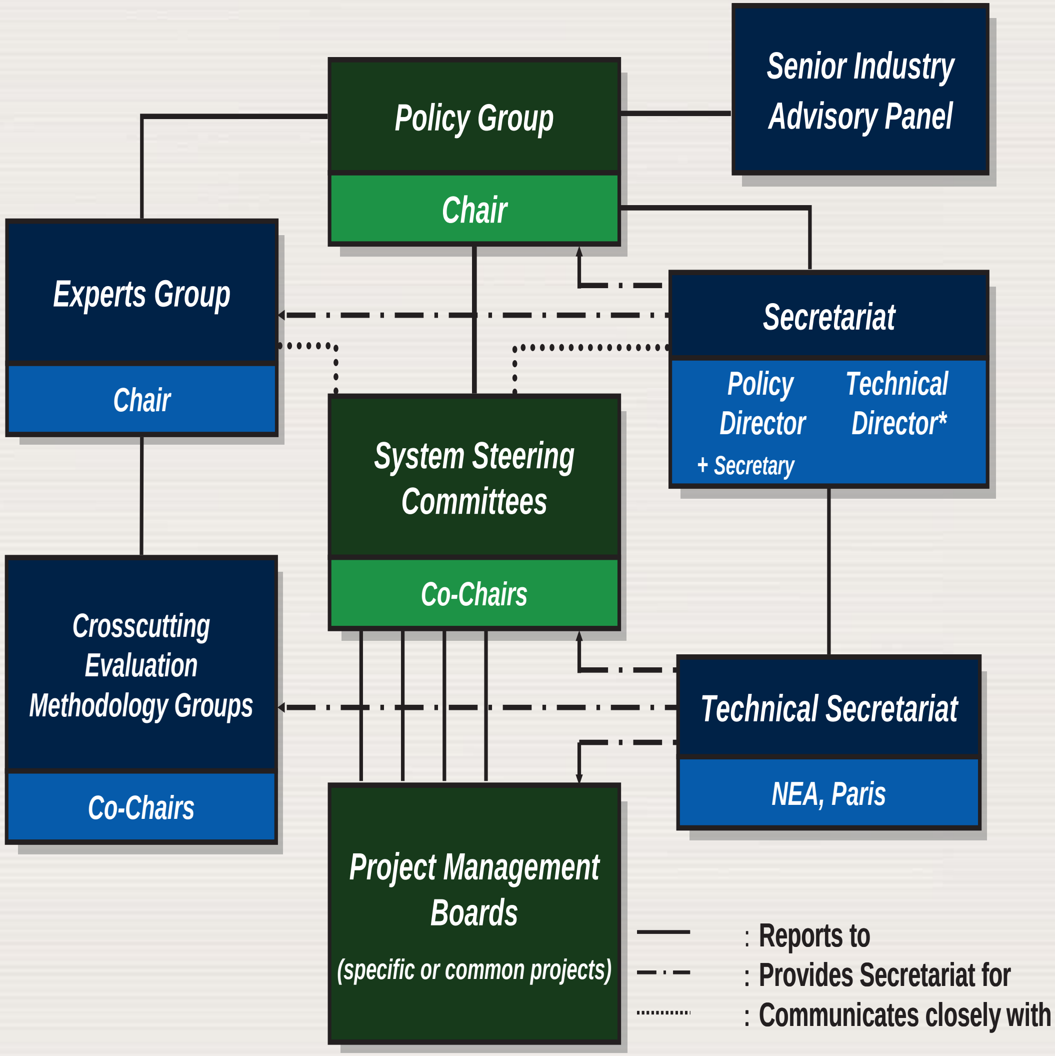


Horizontal Activities
Economics
Safety
Proliferation

GIF (Generation IV International Forum)

Goals and organization:

- ***Sustainability***
  - **resources and waste management**
- ***Proliferation Resistance and Physical Protection***
  - **Not a weapons path; secure against terrorism**
- ***Economics***
  - **Competitive cost and financial risk**
- ***Safety and Reliability***
  - **low risk of core damage, no EP**



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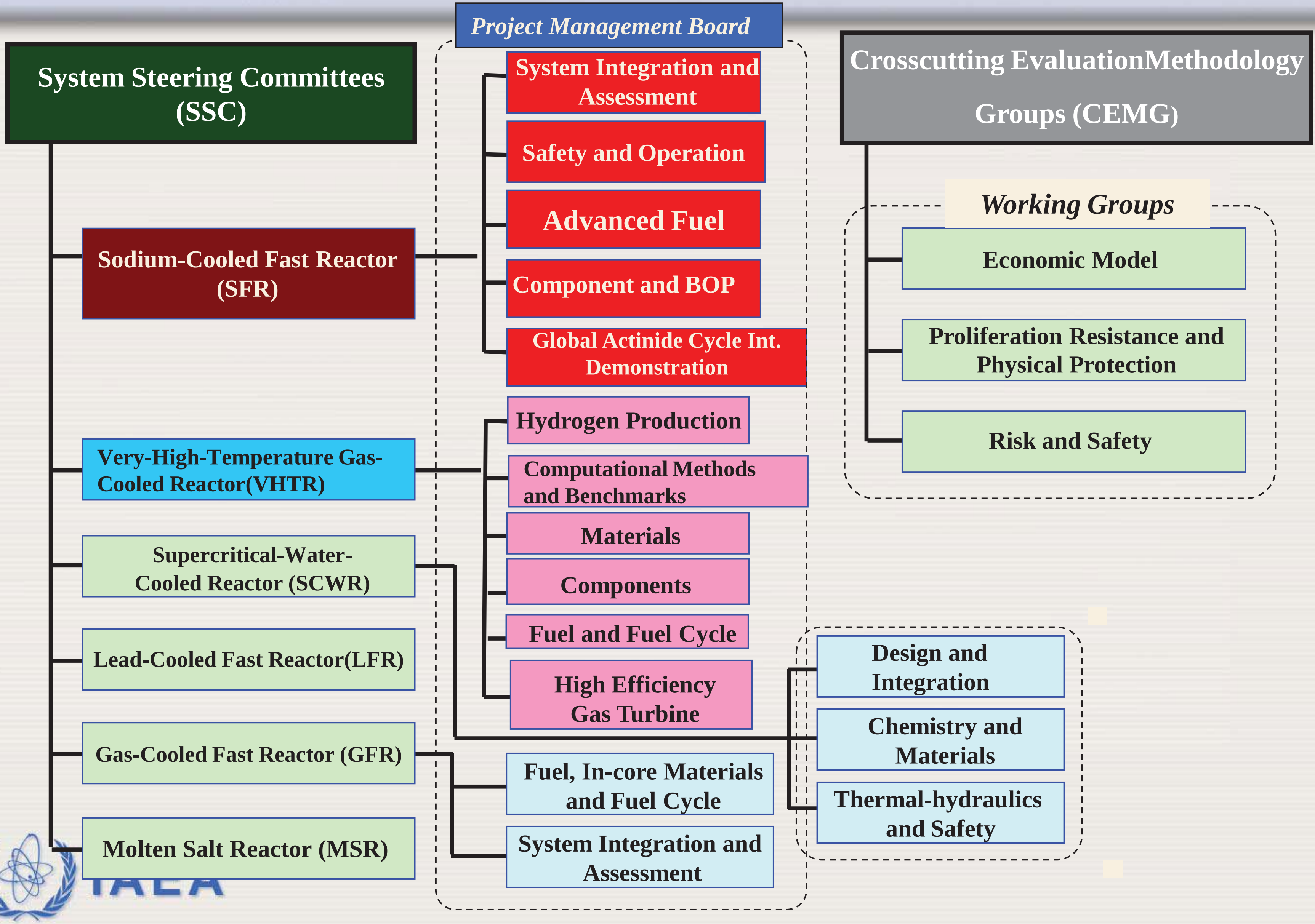
The research and development organization in GIF

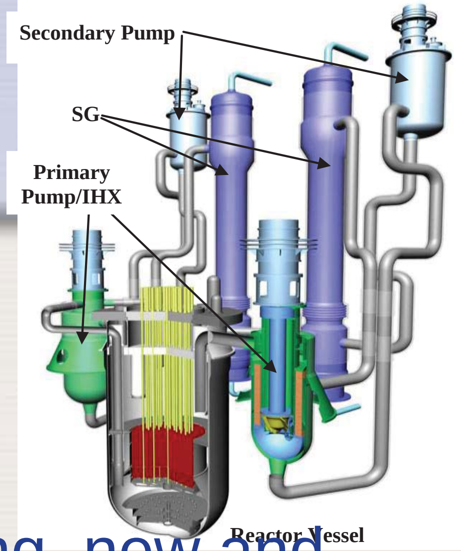


Participating Nations for Generation IV System Developments

Generation IV Systems	Argentina	Brazil	Canada	France	Japan	Korea	South Africa	Switzerland	U.K.	U.S.A.	EU	China	Russia
Sodium-cooled Fast Reactor (SFR)				○	◎	○				○	○	○	(○)
Very-high Temperature Gas-cooled Reactor (VHTR)			○	○	○	○	(○)	○		○	◎	○	
Gas-cooled Fast Reactor (GFR)				◎	○			○			○		
Supercritical-water cooled Reactor (SCWR)			○		○						◎		
Lead-cooled Fast Reactor (LFR)					○					○	○		
Molten Salt Reactor (MSR)				○						○	○		

Note ○: Participating Nations, ◎: Co-chair country, ( ): Observer

Sodium Fast Rector outlook



- Strong national programs and experience
- Goal: Complete performance phase by 2015
- Gather fresh operating experience from existing, new and restarting reactors
- Key technical focus
  - Advanced fuels including actinide recycling
  - Converge safety approach
  - Resolve feasibility issues regarding in-service inspection and repair
  - Energy conversion systems
  - Codes and standards for high temperature application (550 C)

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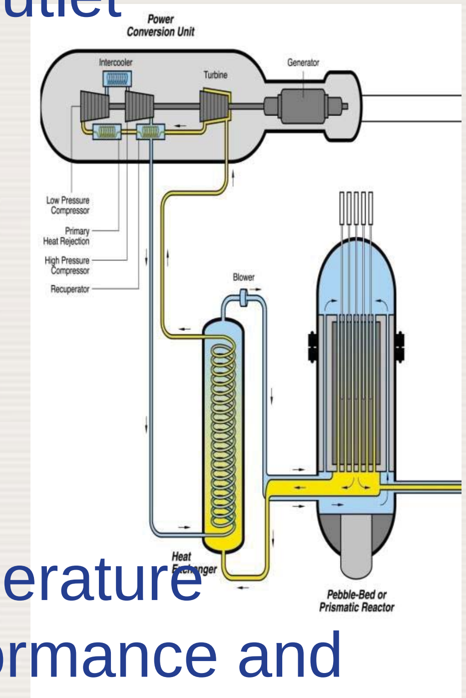
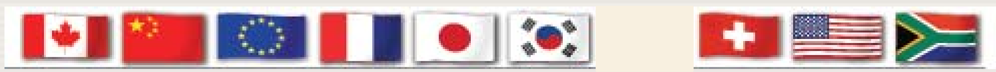
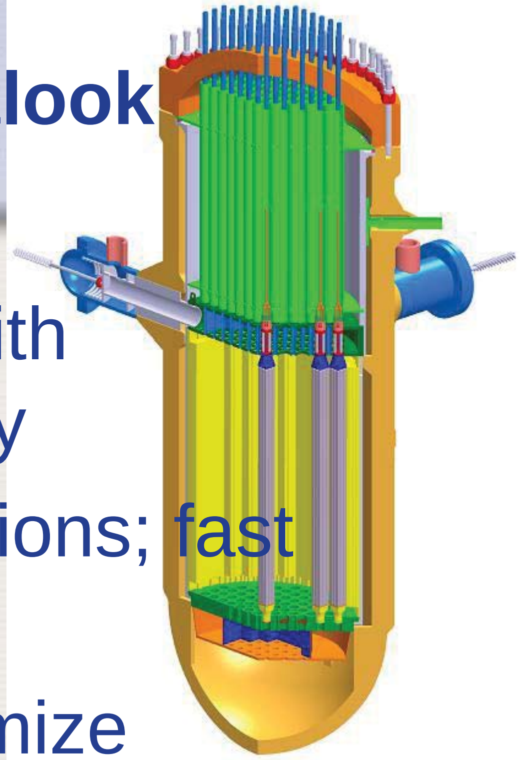
Very High Temperature Reactor outlook

- Strong national programs and GIF arrangements
- Benefits from large national programs with plans for near-term prototype construction of GCRs with outlet temperatures of 750-850 C.
- Goal: Complete viability phase in 2010
- Technical focus
  - TRISO fuel (oxide and oxicarbide)
  - Hydrogen production processes
  - Core and cooling system materials
- Requirements for R&D, set power and temperature requirements (900-1000 C), move into performance and design optimization phase.





Supercritical Water-Cooled Reactor outlook



- Merges GEN-III+ reactor technology with advanced supercritical plant technology
- Pressure vessel and pressure tube options; fast and thermal spectrum options (565 C)
- Goal: Improve knowledge base to optimize designs
- Key technology focus:
  - Materials, water chemistry, radiolysis
  - Thermal hydraulics and property changes around the critical point

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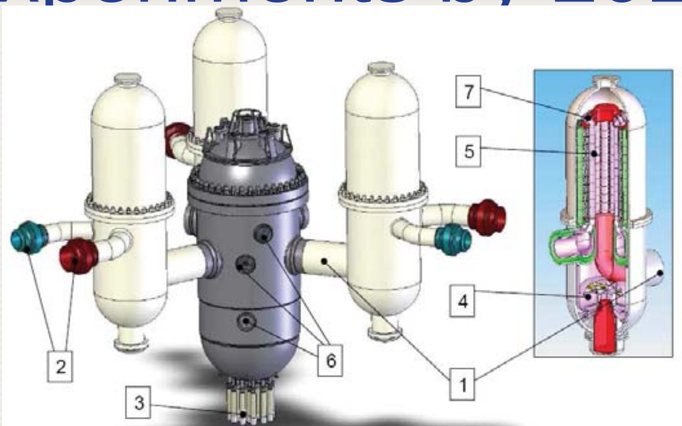
Gas-cooled Fast Reactor outlook

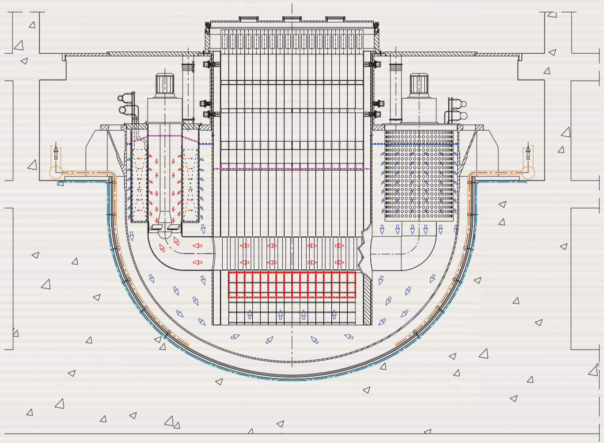
- Limited experience; some benefit from VHTR
- Goal: Complete viability experiments by 2012



- Key technical focus:
  - SiC clad carbide fuel
  - Safety—transient accident analysis
  - Components and materials (850 C)

Lead Fast Reactor outlook

- No formal GIF arrangement; provisional participation
- Goal: Resolve feasibility with respect to components and corrosion control
- Key technical focus:
  - Materials
  - Design features
  - Operating parameters (800 C)



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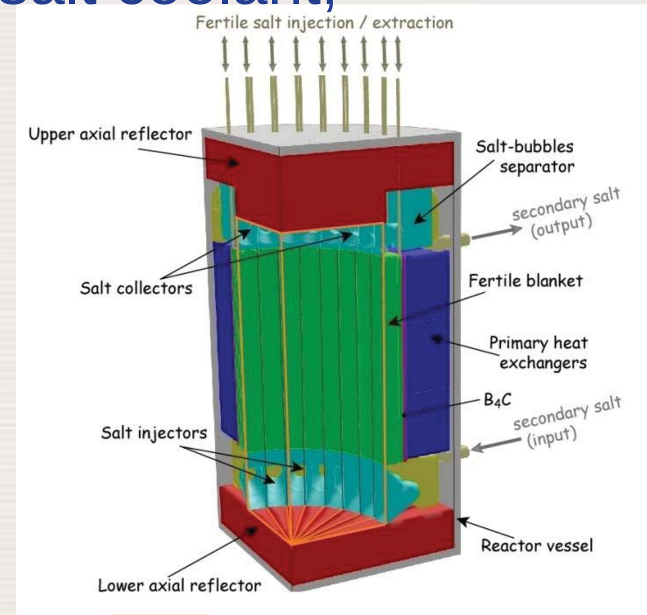
Molten Salt Reactor outlook

700-800 C

- No formal GIF arrangement; provisional participation
- Goal: Develop design features, processing systems and operating parameters within 5 years
- Divergent paths: solid fuel with molten salt coolant; dissolved fuel in molten salt coolant
- Key technical focus
  - Resolve feasibility issues
  - Assess performance of candidate designs
  - Materials
  - In-service inspection
  - On-line salt treatment



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Outlook for horizontal activities

- Economics Methodology Working Group
  - Tool set tested on several systems
  - Available through the NEA
  - Effort is “mature”
- Risk and Safety Working Group
  - Goal is to produce an integrated framework based on PRA
  - Effort is well established; perhaps ripe for re-assessment
- Proliferation Resistance and Physical Protection WG
  - Developing methodology that has been tested on example systems
  - Would benefit from more feedback from GIF Systems

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Summary of GIF Status

- The GIF provides a unique framework for sharing the R&D burden of developing promising nuclear systems through the feasibility, viability and performance phases.
- The systems have some common challenges
  - Higher temperatures for greater efficiency and in some cases additional applications beyond electricity
  - Materials required for the higher temperature and in some cases corrosive environment
  - Advanced components including power conversion
  - Robust, reliable fuel
- Constant pressure to include old/new ideas (thorium, small reactors, etc.) in the light of new developments
- The interface with IAEA/INPRO provides an excellent opportunity to assess allocation of effort.

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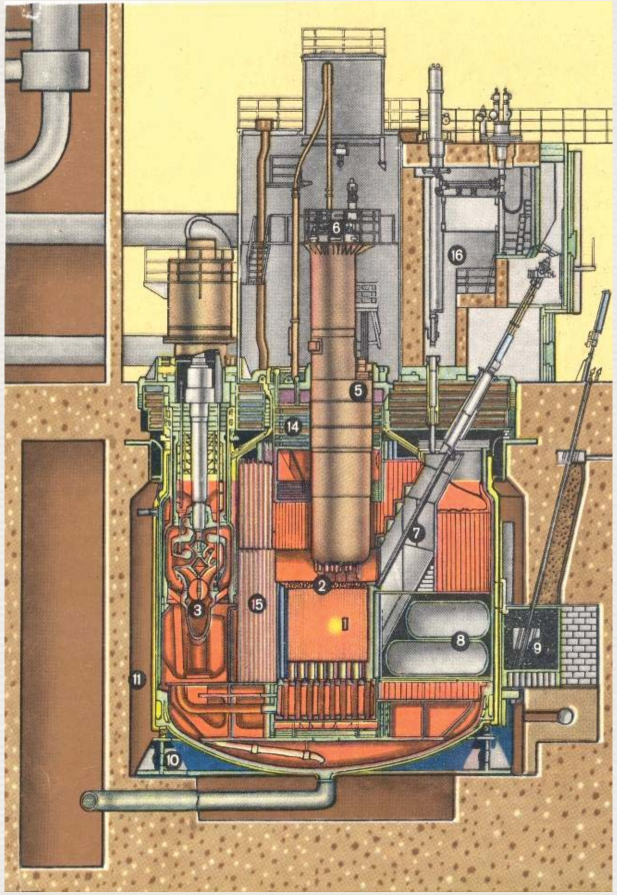
Fast Reactors

BN-600 (Russia)



BN-600 (Russia) BASIC OPERATIONAL PARAMETERS

- *On 08.04.2011, the power unit No.3 of the Beloyarsk NPP with the BN-600 reactor, being the largest operating sodium-cooled fast reactor in the world, celebrated the 31st anniversary since it was connected to the grid*
- *As of 21.12.2010, the BN-600 power unit:*
  - had been in critical condition more than 213 000 hours;
  - had produced about 116.5 billion kW·h of electricity;
  - had operated with average value of the load factor for the period of its commercial operation since 1982 equal to 73.95%.
- *On April 7, 2010, the Beloyarsk NPP received the Rostechnadzor's license for lifetime extension of the BN-600 power unit up to March 31, 2020.*
- *On April 13, 2011, the BN-600 power unit was shutdown in accordance with scheduled and standard preventive maintenance works and refueling are performed now.*
- *In 2010:*
  - 3932.61 million kW·h of electricity had been produced,
  - 256 Tcal of heat had been delivered for heat consumers from power unit's collectors.
  - NPP load factor was equal to 74.82%.

BN-800 (Russia) under advanced construction

Bird's-eye view of the BN-800 power unit

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BN-800 (Russia) under advanced construction

Bird's-eye view of the reactor compartment of the main building

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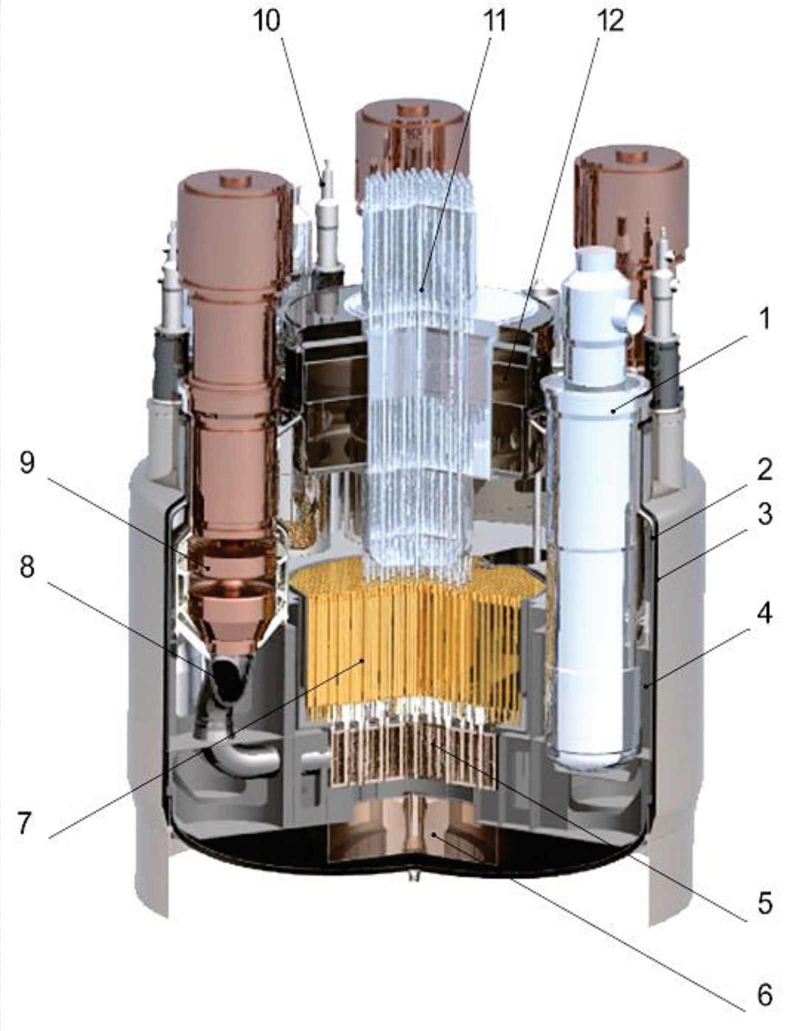
BN-800 (Russia) under advanced construction

Bird's-eye view of the turbine hall

BN-1200 (Russia)



1 – IHX;
2, 3 – main and secondary pressure vessels respectively;
4 – supporting structure;
5 – inlet plenum;
6 – core debris tray;
7 – core;
8 – pressure pipeline;
9 – MCP-1;
10 – refueling mechanism;
11 – CRDM;
12 – rotating plugs.

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PFBR (India)

Nuclear island connected building was completed and the roof of reactor containment building was concreted partially

Overall view of NICB

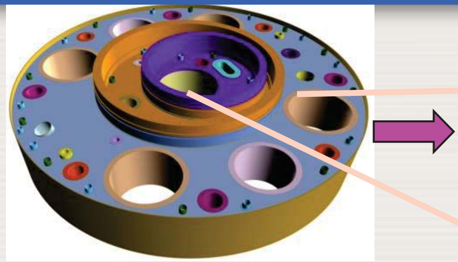
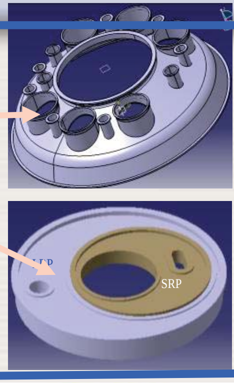
Lowering of Caisson in sea bed

- TG Deck concreting
- The raft of sea water pump house
- 3 stages concreting of volute pump.

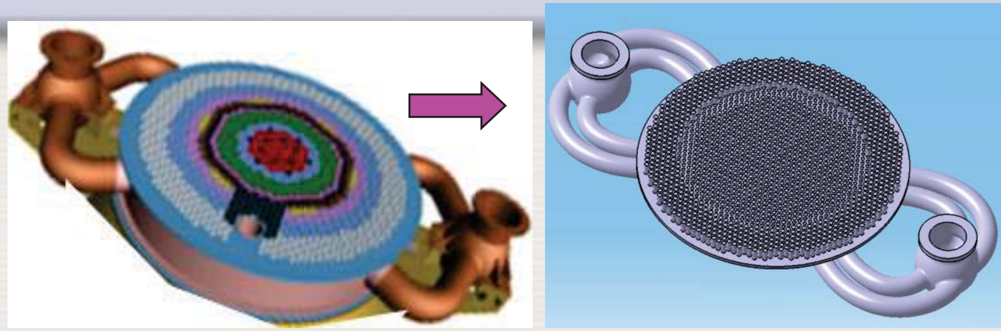
Overall view of Turbine building

33m girders & trolley of 280/85 T EOT in RCB

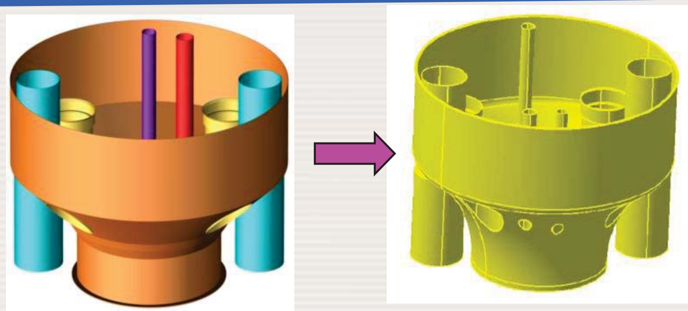
Design Evolutions: PFBR to CFBR (India)

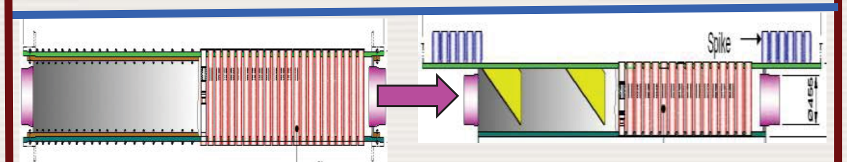
Box to plate concept



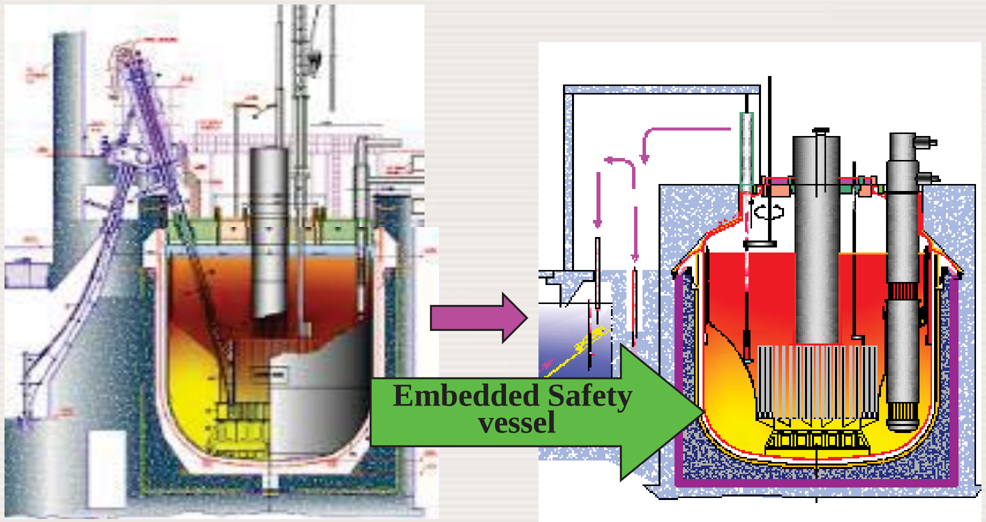
Four to eight primary pipes



Cone with torus to single torus



Large plenum to smaller plenum
Bolted structure to welded structure



Elimination of Inclined fuel transfer machine

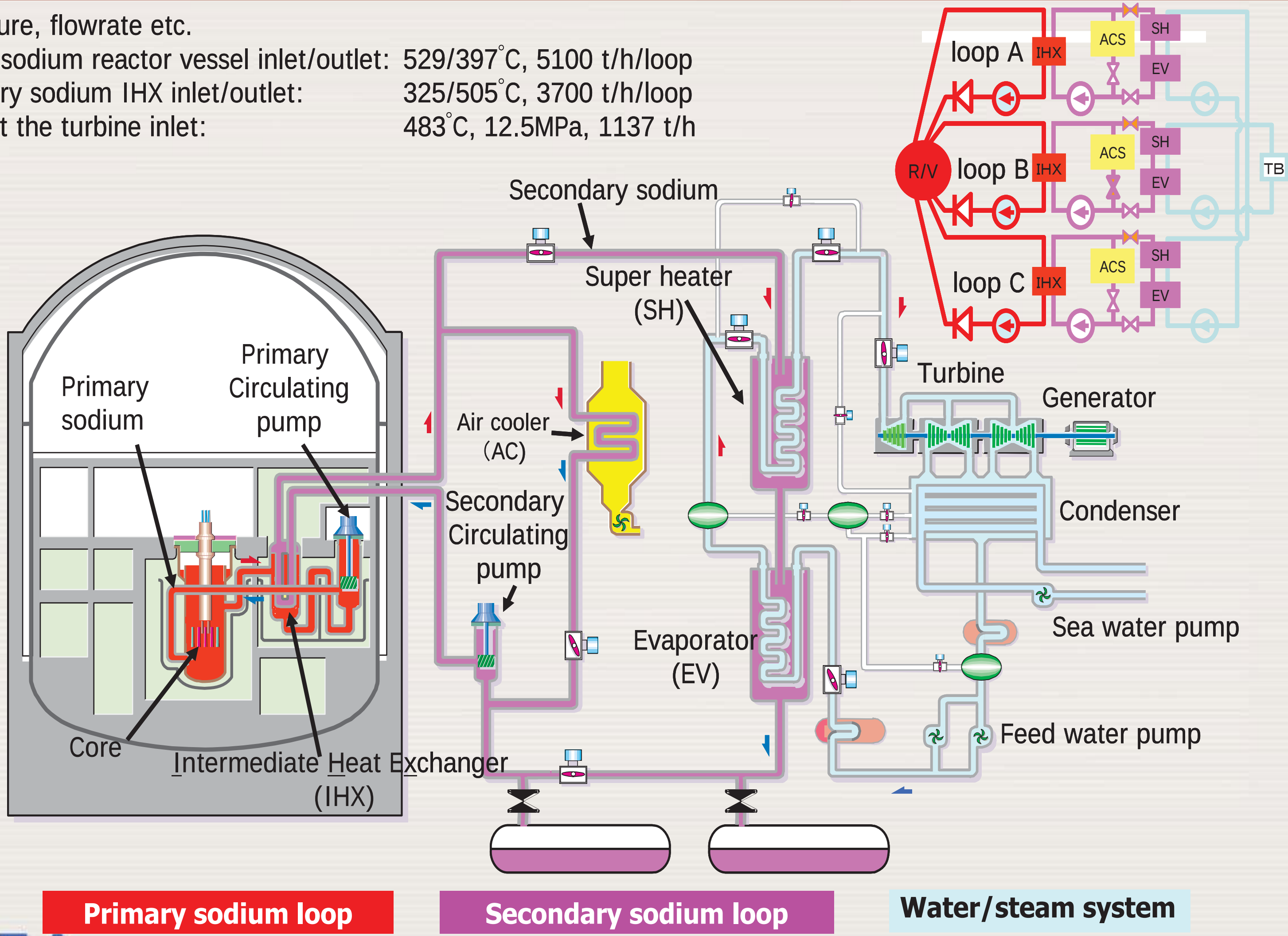
Material inventory reduction~ 25% , Simplified fuel handling scheme, Reduced manufacture time, Enhanced safety

Schematic of Monju (Japan)

Electricity Output : 280MWe (714MWt), Sodium Coolant, MOX Fuel Core

Temperature, flowrate etc.

Primary sodium reactor vessel inlet/outlet:	529/397°C, 5100 t/h/loop
Secondary sodium IHX inlet/outlet:	325/505°C, 3700 t/h/loop
Steam at the turbine inlet:	483°C, 12.5MPa, 1137 t/h



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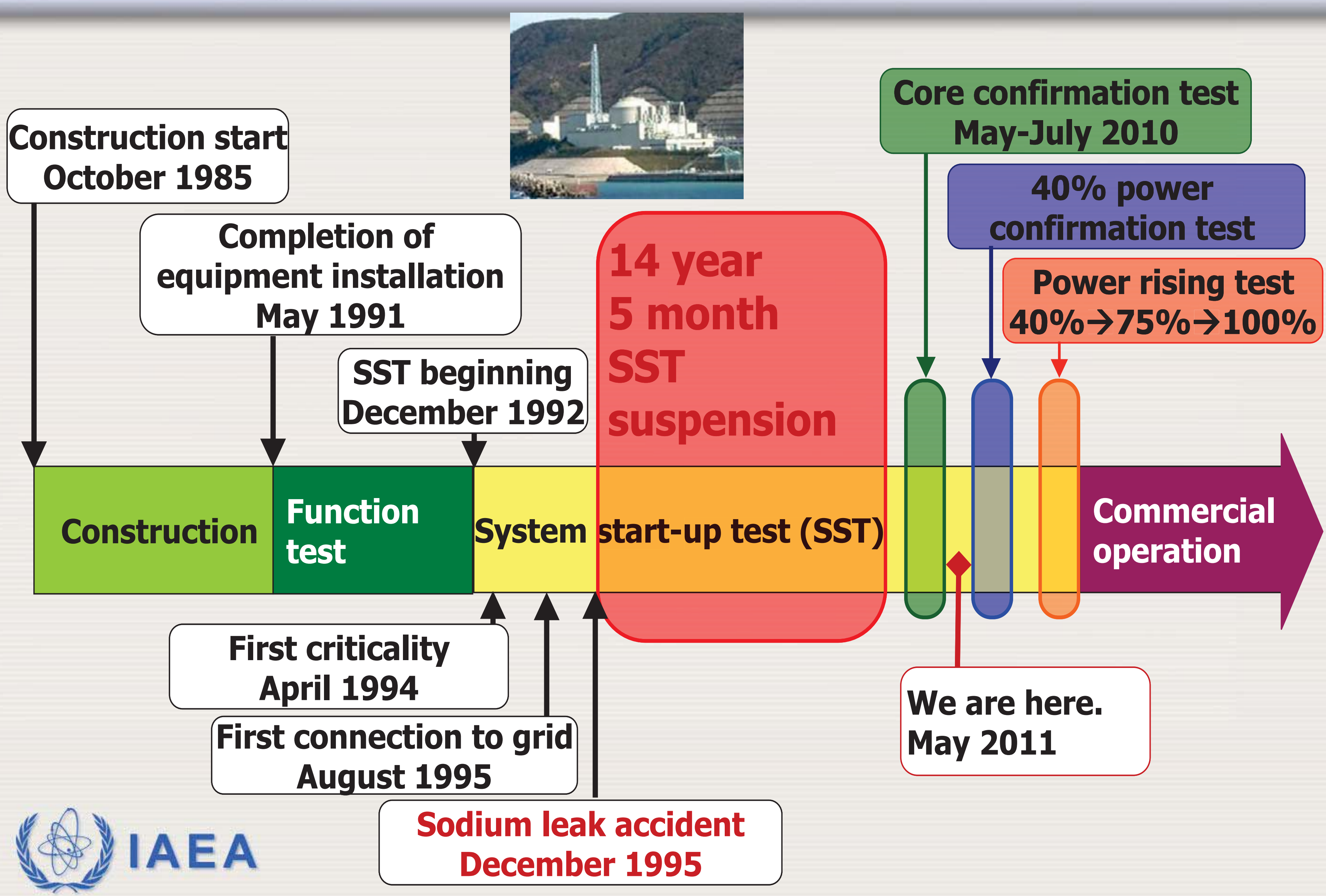
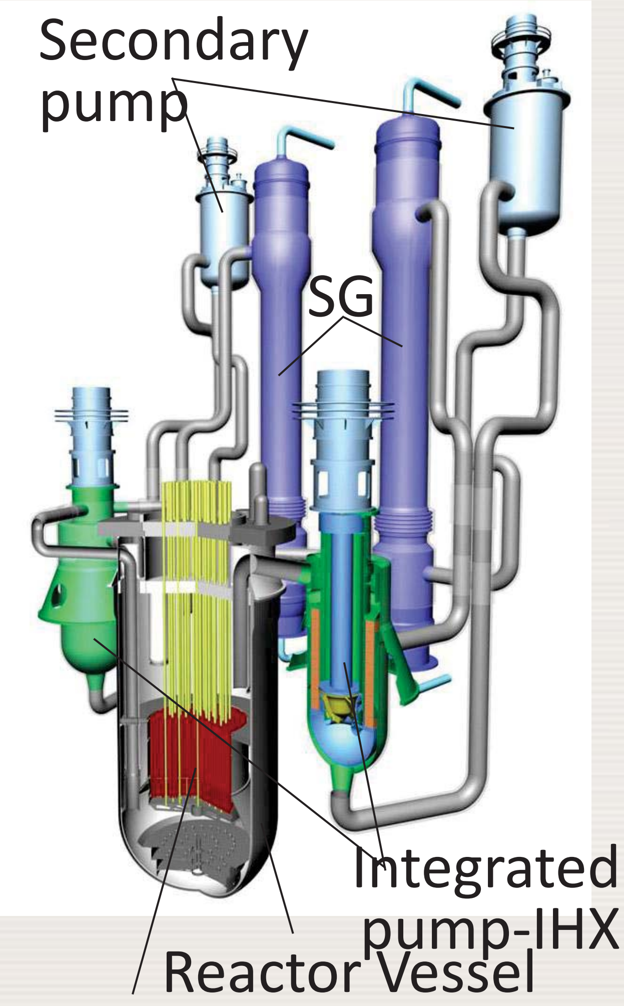
Progress of Monju



Japan Sodium-cooled Fast Reactor (JSFR)



Items	Specifications
Output	3,530MWt / 1,500MWe
Number of loops	2
Primary sodium temperature	550 /395 degree C
Secondary sodium temperature	520 / 335 degree C
Main steam temperature and pressure	497 degree C 18.7 MPa
Feed water temperature	240 degree C
Thermal efficiency of plant	Approx. 42%
Fuel type	TRU-MOX
Breeding ratio	Break even (1.03) ~ 1.2
Cycle length	26 months or less, 4 batches

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CEFR (China)

Preparation of Site

1998.10

FCD

2000.5

Main building finished

2002.8

Reactor block installation finished

2008.12

Commissioning of Phase A ended

2009.8

Main parameters of CEFR

Parameter	Unit	Value	Parameter	Unit	Value
Thermal Power	MW	65	Primary Circuit		
Electric Power, net	MW	20	Number of Loops		2
Reactor Core			Quantity of Sodium	t	260
Height	cm	45.0	Flow Rate, total	t/h	1328.4
Diameter Equivalent	cm	60.0	Number of IHX per Loop		2
Fuel		(Pu, U)O_2 (first loading is UO_2)	Secondary Circuit		
Linear Power max.	W/cm	430	Number of Loop		2
Neutron Flux	$n/cm^2 \cdot s$	3.7×10^{15}	Quantity of Sodium	t	48.2
Bum-up, first load max.	MWd/t	60000	Flow Rate	t/h	986.4
Inlet/outlet Temp. of the Core	℃	360/530	Tertiary Circuit		
Diameter of Main Vessel(outside)	m	8.010	Steam Temperature	℃	480
Design Life	A	30	Steam Pressure	MPa	14
			Flow Rate	t/h	96.2

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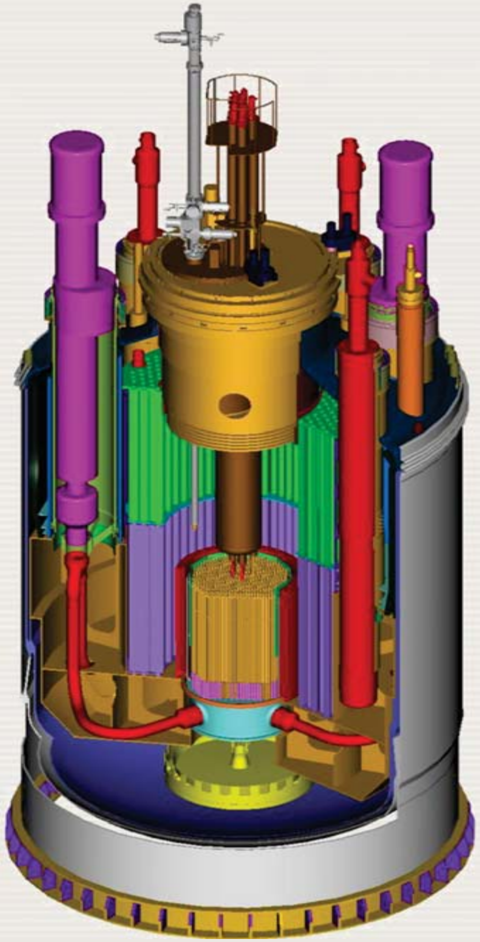
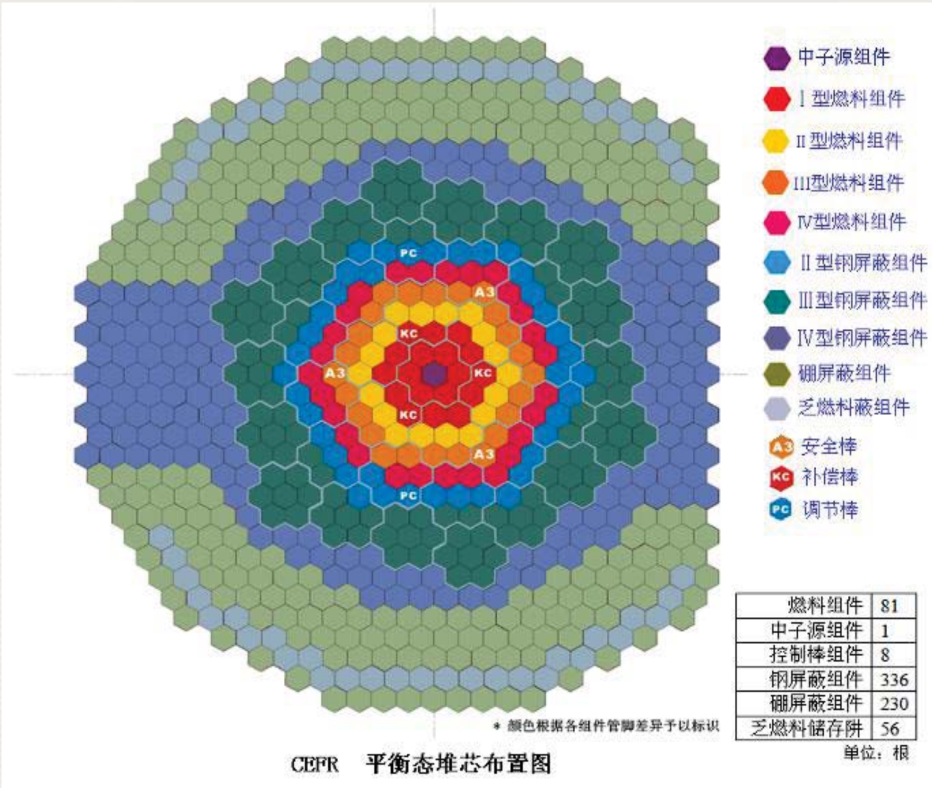
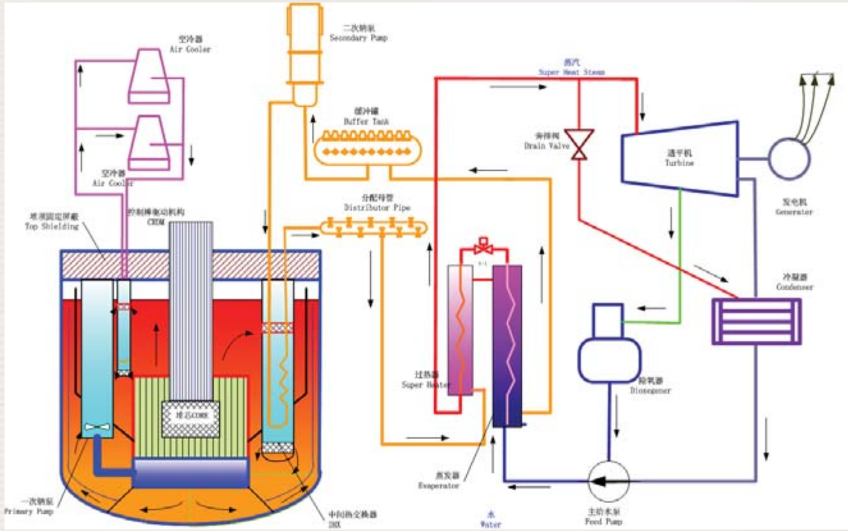
Reactor block and reactor core





CFR-1000 (China)

- 2500MWth, ~1000MWe
- MOX fuel
- BR=1.2
- Sodium as coolant
- Na-Na-H_2O loops with 3 circuits of primary and secondary loop
- One turbine
- Negative feedback
- Confinement
- CDR<10^{-6}
- Life>40y



CFR-1000 scheme

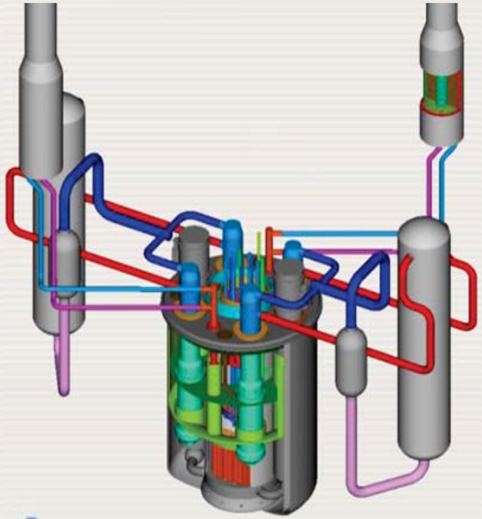
Main parameters of CFR-1000

Parameters	Value
Thermal power	2500 MWt
Electricity power	1000 MWe
Efficiency	40%
Loading factor	80%
Life time	>40y
Fuel	MOX
Coolant	Sodium
Primary loop type	Pool
Burn up (average)	66 MWd/kg
Burn up (maximum)	100 MWd/kg
Breeding ratio	1.2
Number of fuel subassembly (inner/out zone)	184/132
Enrichment (Inner/out)	15.3%/19%
Liner power (maximum)	29.3/49 kW/m
Total number of fuel SA	316
Rods number per SA	271

Parameters	Value
Number of control rods	30
Number of safety rod/shim rod/regulatory rod	6/19/2
Passive control rods	3
Blanket SA	255
Reflector SA	426
Spent fuel SA storage trap	381
Inner diameter of MV	14000 mm
Thickness of MV	30 mm
Sodium temperature (core inlet/outlet)	354/547℃
Sodium temperature (IHX inlet/outlet)	544/352℃
Primary loop flow rate	10100 kg/s
Loop number (Primary/secondary)	3/3
Pump number per loop	1/1
IHX number per loop	2

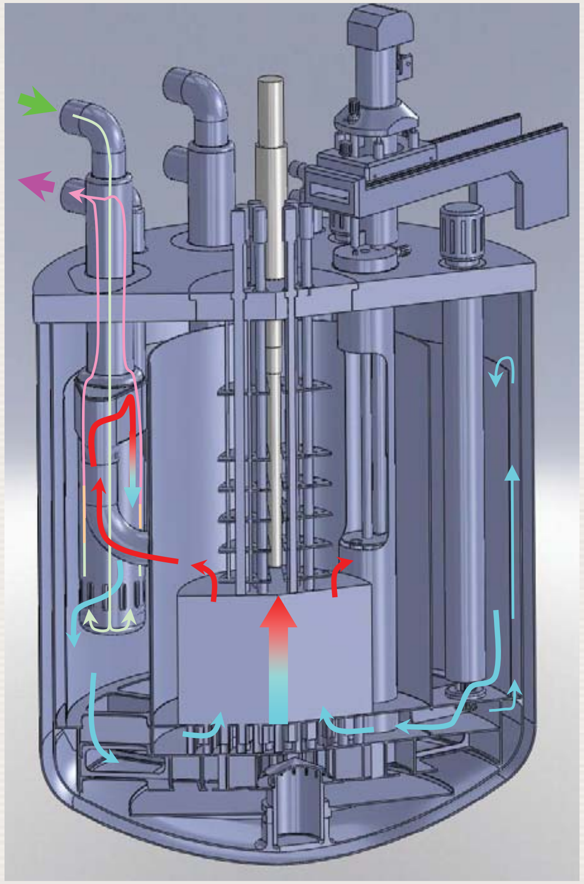
Kalimer – 600 (Republic of Korea)

- Reactor type : pool type
- Plant size : 600MWe
- Plant design lifetime : 60 years
- Design basis earthquake (SSE: 0.3g)
- Initial core : U-Zr metal fuel
- Reloading core : U-TRU-Zr metal fuel



- Plant thermal efficiency : Net > 38%
- Plant availability ≥ 70 %
- Refueling interval
  . U-Zr initial core : ≥ 6 months
  . TRU burner core : ≥11 months
- Spent fuel storage capacity in RV
  ≥ 1.5 cycle discharge
- 100 % off-site load rejection w/o a plant trip
- Safety grade diesel generator

ASTRID (France): Advanced Sodium Technological Reactor for Indusrial Demonstration (approx. 600 MW_e)



Preconceptual study by AREVA

- French government (CEA) led project: national loan 650 M€
- Industry participation - EDF operations feedback from previous reactors, utility requirements and design, safety (all aspects)
- Industry participation - AREVA : from pre conceptual to detailed design studies for the Nuclear Steam Supply System
- Other industrial partners are invited by CEA
- Current project phase: design choices to move from pre conceptual design to conceptual design coupled to ongoing R&D
- and discuss with the regulator the safety orientations (2012)

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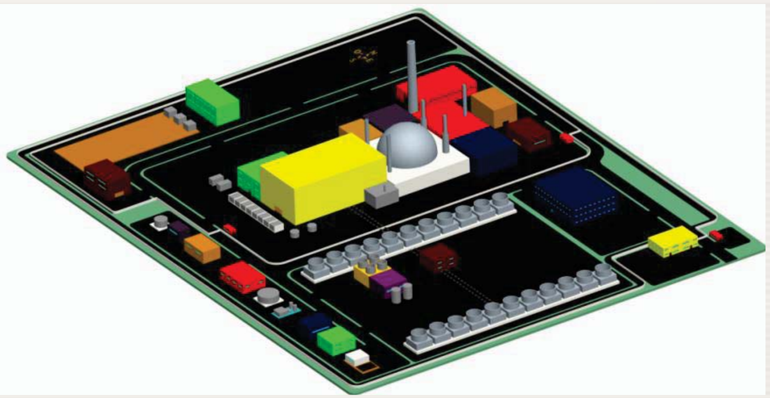
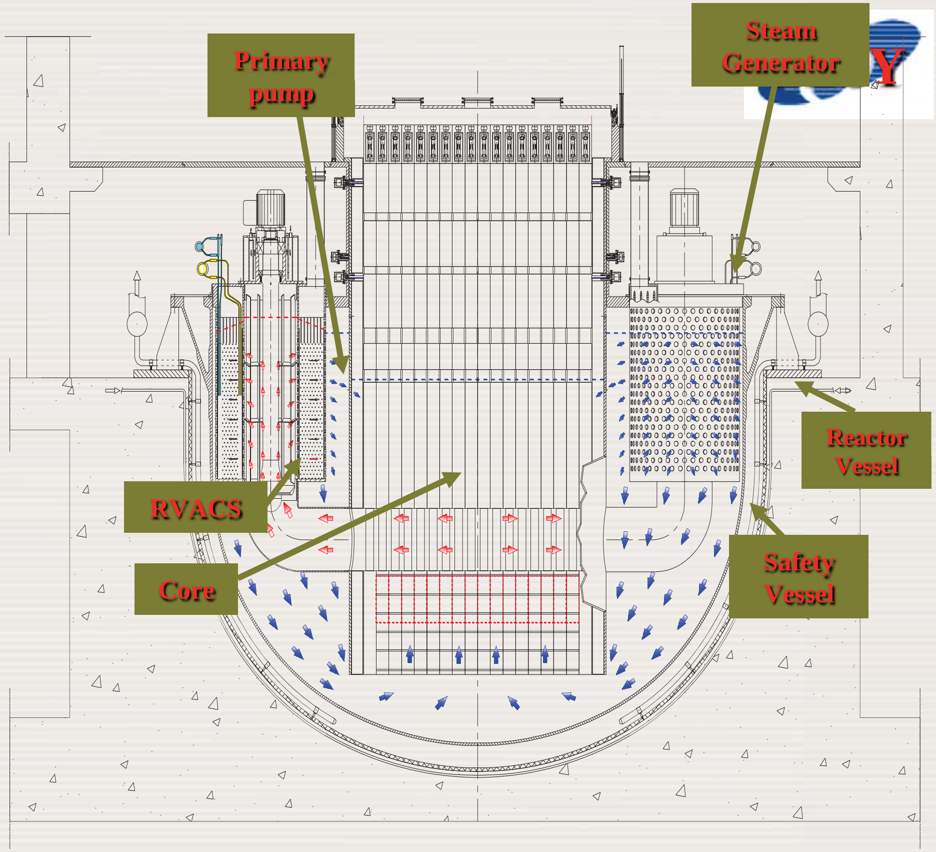
ELSY - European Lead-cooled System





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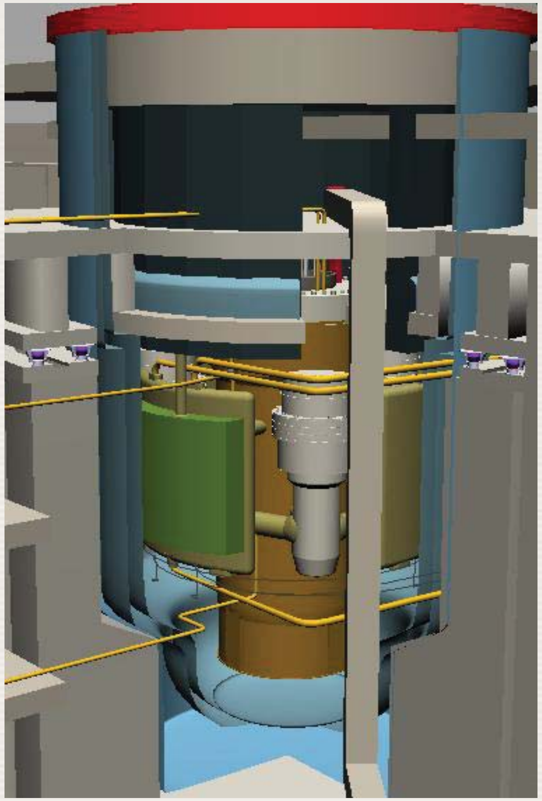
Main parameters of ELSY (now LEADER)

	ELSY
Power, MWe	600
Thermal efficiency %	42
Primary coolant	Pure lead
Primary coolant circulation (at power)	Forced
Primary coolant circulation for DHR	Natural
Core inlet temperature °C	~ 400
Core outlet temperature °C	~ 480
Fuel	MOX with and without MA
Neutron spectrum	Fast
Fuel pin diameter, (mm)	10.5
Fuel cladding temperature (max) °C	~ 550
Active core dimensions Height/ equivalent diameter, (m)	0.9/4.32
Fuel column height. (mm)	900
N° Fuel Assemblies (FA)	162
FA geometry	Open
FA pitch, (mm)	294
N° fuel pins / FA	428
Fuel pins pitch at 20° C, (mm)	13.9 square
Enrichment, (%wt HM)	14.54/17.63/20.61 Pu, three radial zones
Power conversion system working fluid	Water-superheated steam at 18 MPa, 450° C
Primary/secondary heat transfer system	Eight Pb-to-H2O SGs

PEACER-300 (SNU, RoK)



3D CAD Drawing of PEACER-300

Parameter	Value
Power (MWe/MWt/efficiency)	300/850/35.3
Core outlet/inlet coolant temperature (°C)	400 / 300
Primary coolant	Pb-Bi (45-55 Wt. %) eutectic alloy
Primary system cover gas pressure (atm)	1.0
Primary coolant flow rate (kg/sec)	58 000
Average power density, MWt/ m^3	205
Average discharge burnup (GWd/t)	76.6
Fuel composition	U-TRU-Zr (57-32-11 Wt. %) 3 TRU Enrichment Zoning Smeared Density : 67%
Fuel assembly	PWR type open lattice (square array without wrapper)
Cladding material	HT-9
Fuel/coolant volume fraction	0.159/0.677
Fuel assembly lifetime (year)	3
Fuel pin diameter, cm	0.832
Fuel rod pitch to diameter ratio	1.44
Active core dimensions: height / diameter, m	0.50/3.82
Energy converter / balance of plant	Superheated steam cycle
Fuel cycle strategy	Full actinide recycle using pyrochemical partitioning facility
In core management	3 Batch annual reload
Passive safety	Negative temperature coefficient and Pb-Bi natural circulation
Emergency decay heat removal	Guard vessel cooling by natural circulation of air; always in effect and enhanced by LBE flooding during emergency
Design seismic acceleration (g) (horizontal, vertical)	3D Seismic Isolators (0.3, 0.2)
Electric capacity factor	90%

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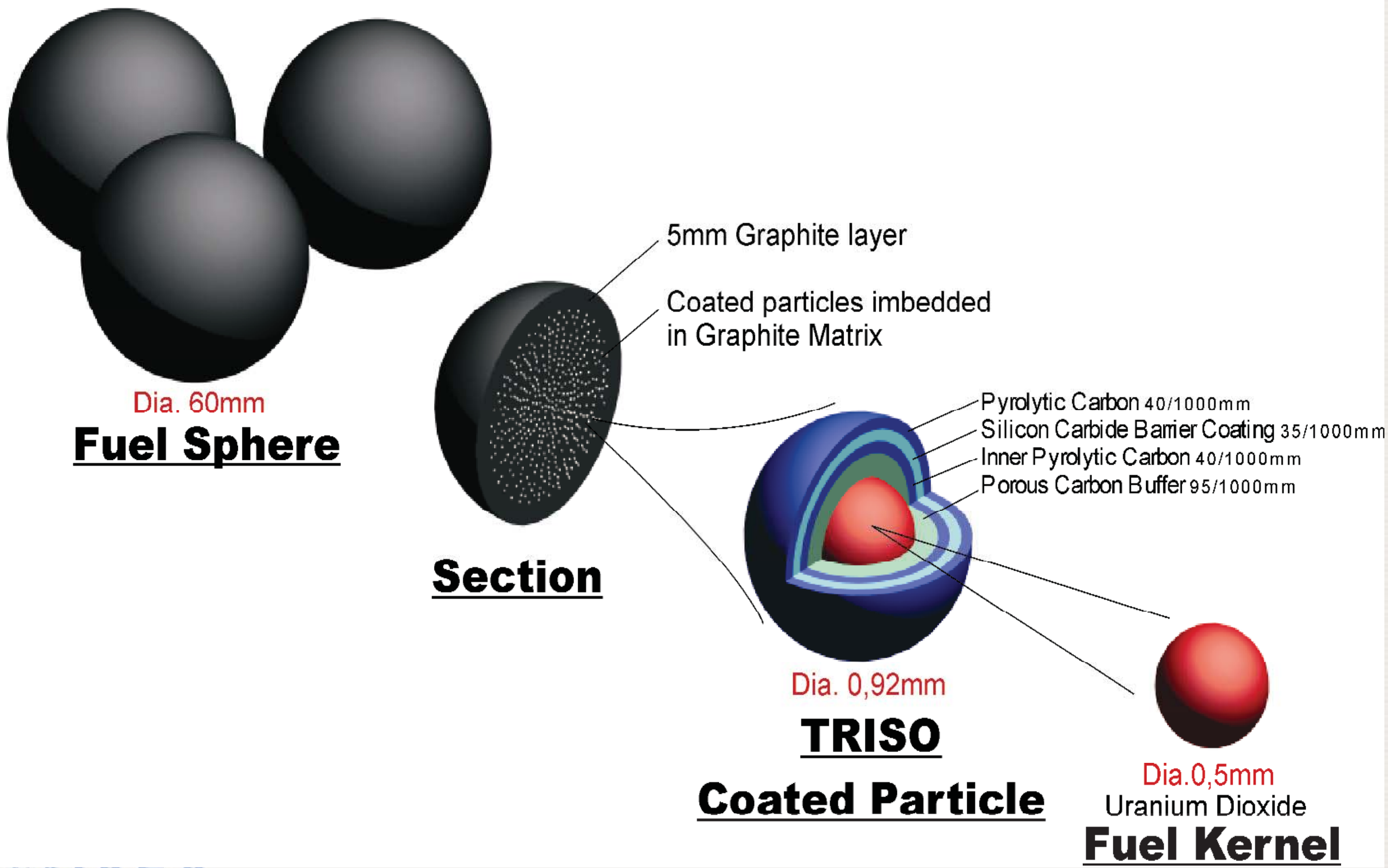
Gas cooled Reactors

SALIENT FEATURES



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OVERVIEW

EXPERIMENTAL REACTORS

DEMONSTRATION OF
BASIC HTGR TECHNOLOGY

DRAGON
(U.K.)
1963 -1976

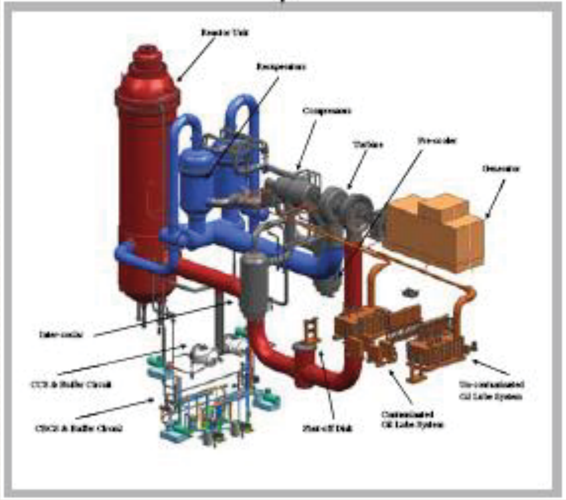
AVR
(FRG)
1967 - 1988

HTTR
(Japan)
1998 - Present

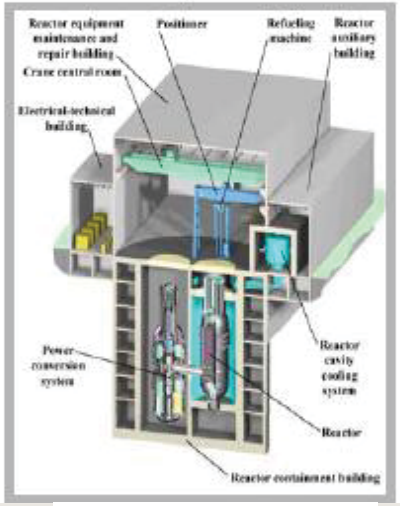
PEACH BOTTOM 1
(U.S.A.)
1967 - 1974

FORT ST. VRAIN
(U.S.A.)
1976 - 1989

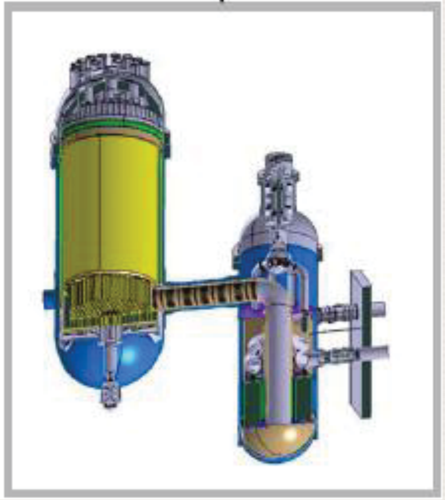
THTR
(FRG)
1986 - 1989



PEBBLE BED MODULAR REACTOR
PBMR



MODULAR HTGR CONCEPT
GENERAL ATOMICS



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European Gas Fast Reactor Prototype: Allegro

- Longer term alternative technology of fast spectrum reactors for sustainability
- Able to deliver high temperature heat for industrial processes (H_2 production)
- After an R&D program to consolidate preconceptual design options, ALLEGRO demonstrator (70-100 MW_e) to be launched in 2012
- Central European ALLEGRO Consortium: Czek Republic, Slovakia, Hungary
- Ongoing process for Site selection and site permit (Eastern Europe)
- Euratom GoFastR collaborative project in support (22 partners)

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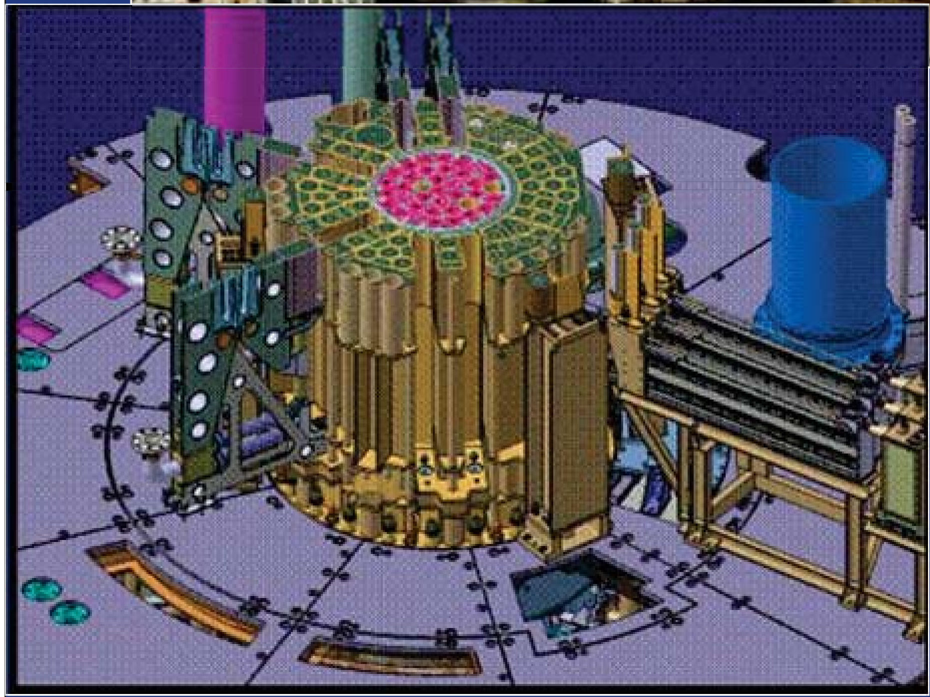
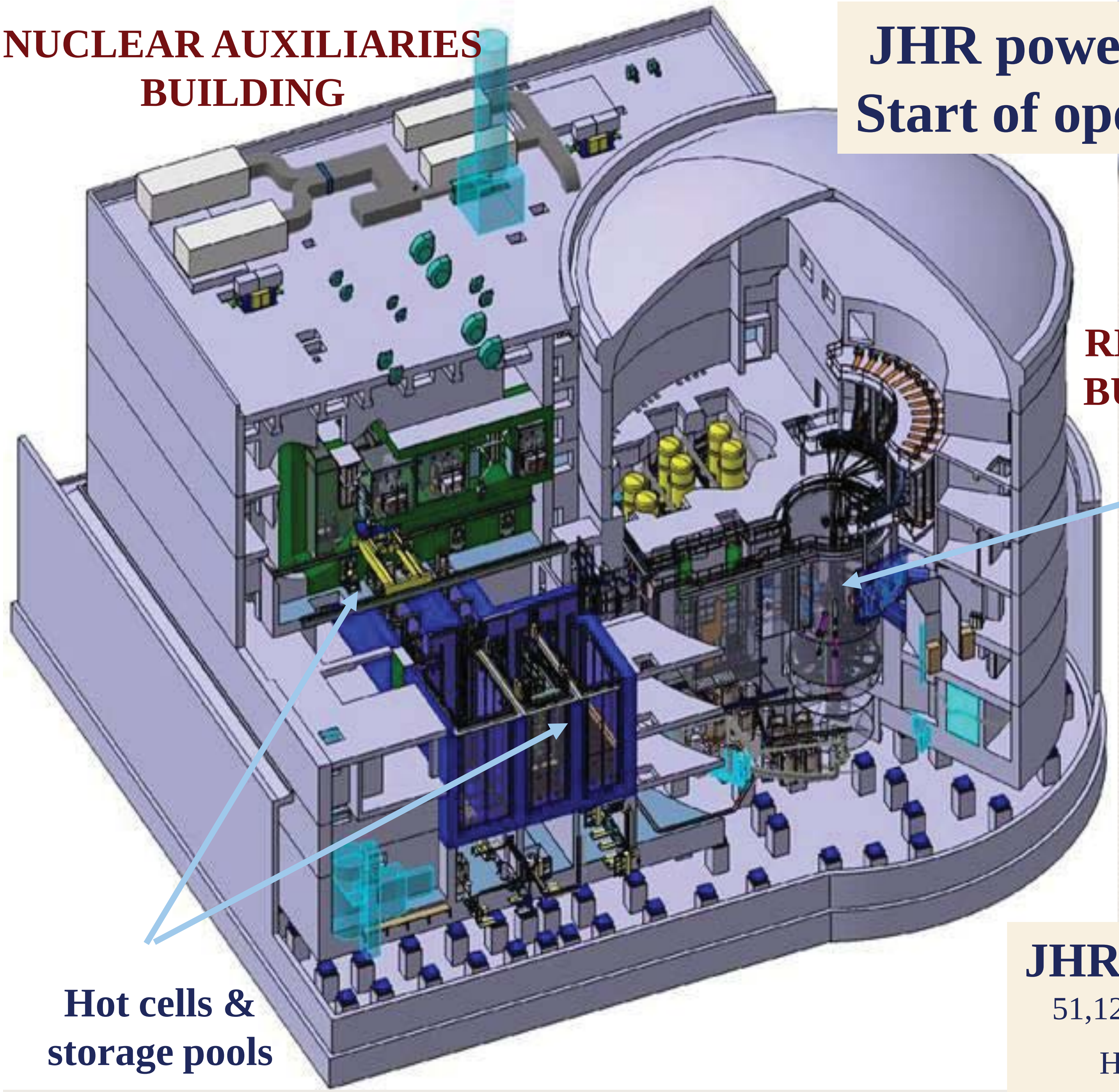
Material Test Reactors

Jules Horowitz Reactor (*France*) a new European Testing Reactor



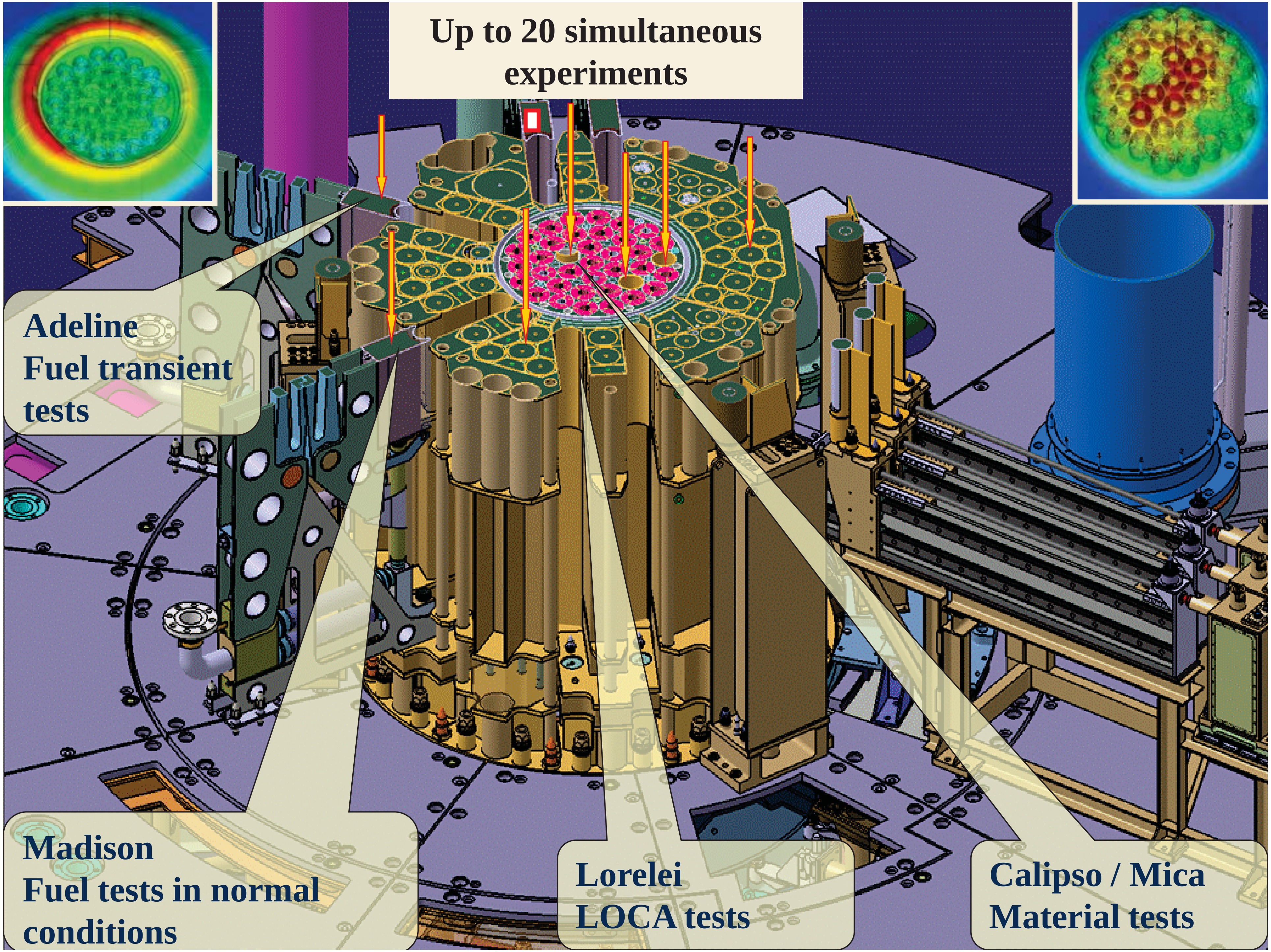
JHR power = 100MW
Start of operation 2014



JHR characteristics

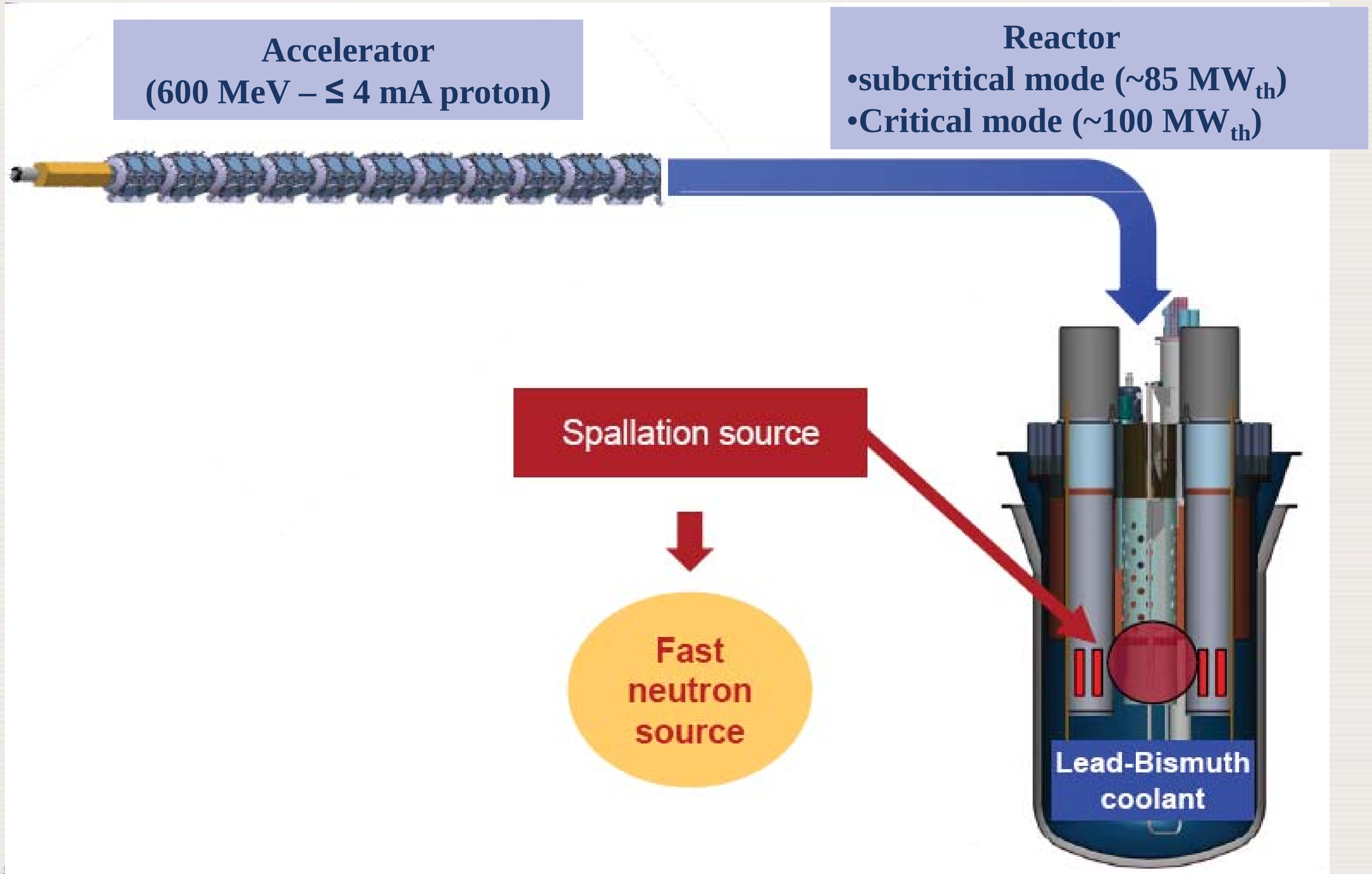
51,12m x 46,75m + Φ36.6m

H 34,4m + H44,9 m



Up to 20 simultaneous experiments
Adeline
Fuel transient tests
Madison
Fuel tests in normal conditions
Lorelei
LOCA tests
Calipso / Mica
Material tests

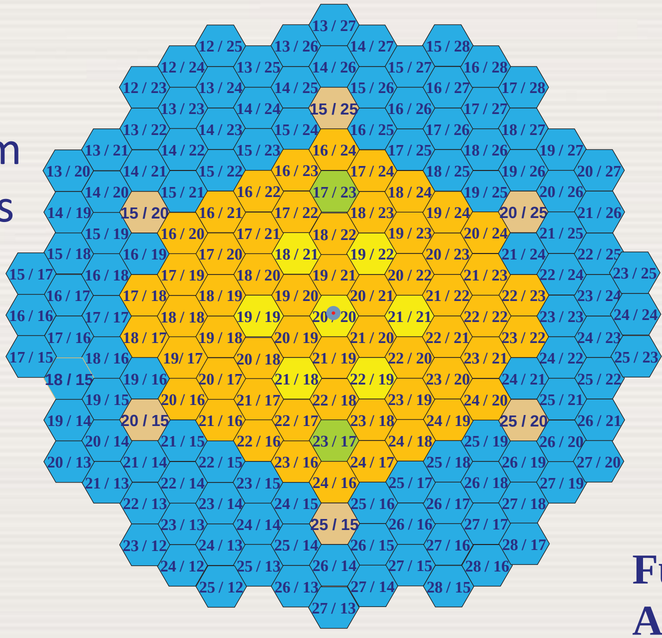
MYRRHA by SCK.CEN (Belgium)

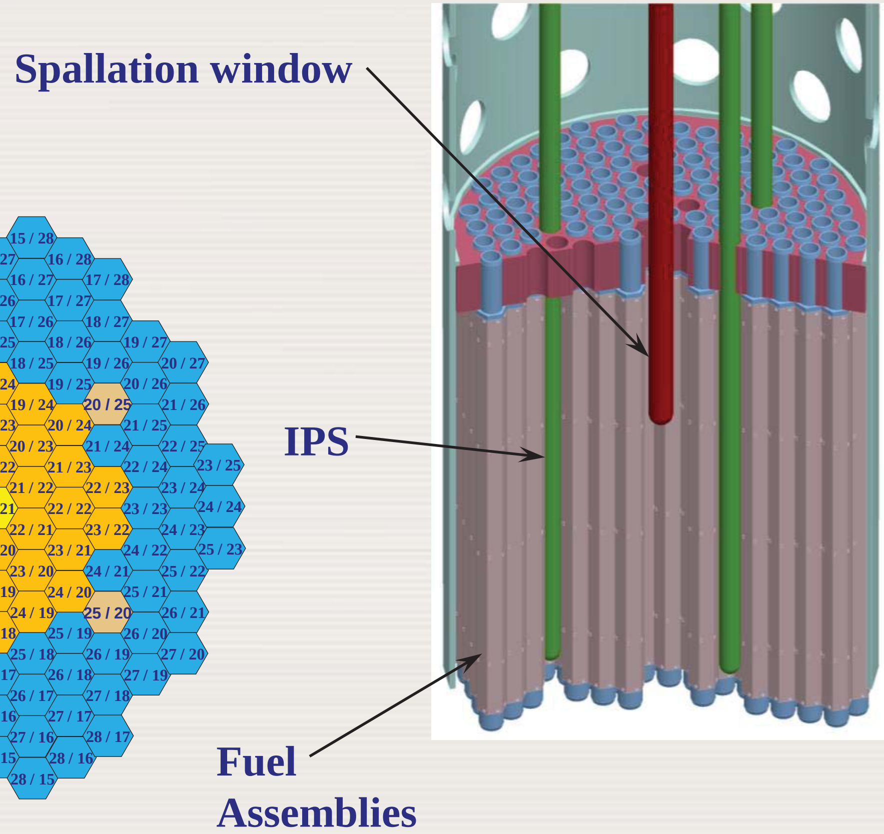


MYRRHA Core

- $k_{eff} \approx 0.95$ (ADS mode)
- 30-35 % MOX fuel
- 7 IPS positions

56 FAs
7 IPS
2+6 Abs
116 Dum
187 S/As

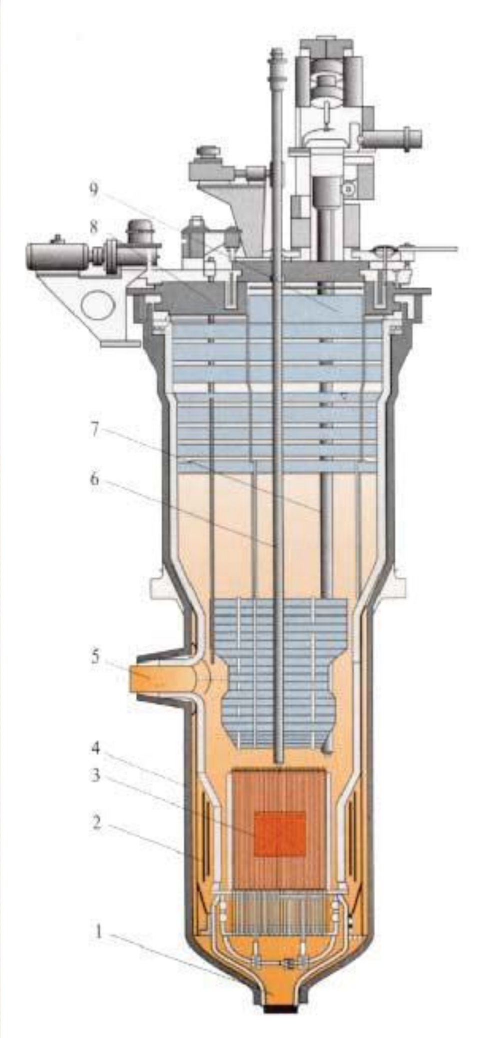




BOR-60 (Russia) BASIC OPERATIONAL PARAMETERS

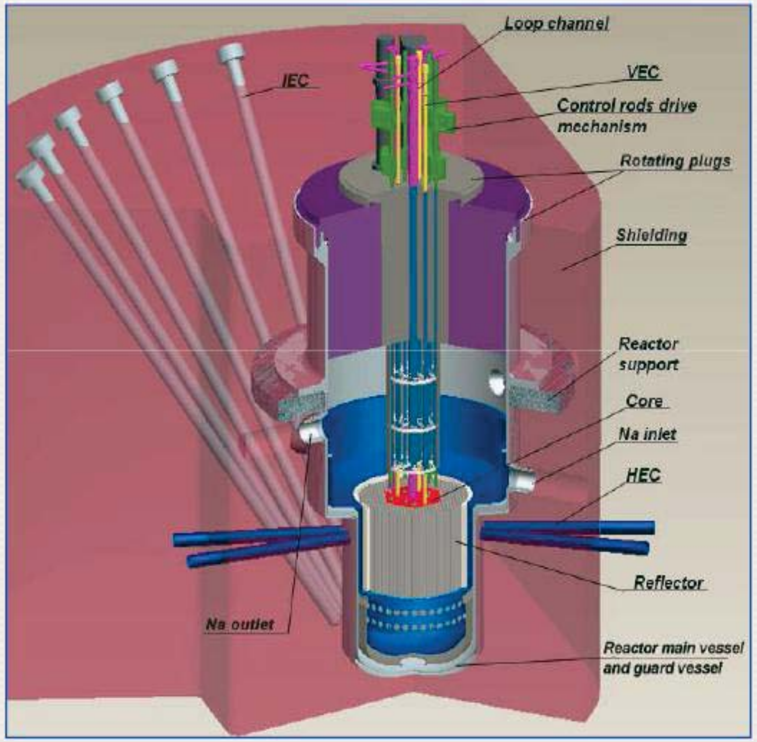
- *The BOR-60 reactor is used for:*
  - Material tests;
  - Isotopes production (nickel-63, strontium-89, gadolinium-153);
  - Tests of the various equipments of fast reactors;
  - Heat and electricity production.
- *BOR-60 reactor facility is in operation more than 41 years.*
- *In December 2009, Rostechnadzor has issued a license to RIAR for further operation of the BOR-60 reactor facility up to 31.12.2014.*

INDICES OF BOR-60 REACTOR OPERATION IN 2010



Index	Value
Time of reactor operation on power exceeding minimum controlled level, h	5245
Reactor load factor	0.6
Max reactor power, MW	53
Energy output: heat, MW·h electricity, MW·h	 251805 38480
Time of SG operation: SG-1, h SG-2, h	 5173 5173
Heat delivery to consumers, Gcal	47451

MBIR (Russia)



Lay-out of the MBIR reactor vessel and its experimental channels

- *Na-cooled Research Fast Reactor aimed at in-pile tests of new types of fuel, structural materials and various FR coolants (including Pb and Pb-Bi)*
- *Start-up of MBIR is scheduled in 2019.*

BASIC CHARACTERISTICS OF THE MBIR REACTOR

Parameter	Value
Thermal power, MW	~150
Electric power, MW	~40
Maximum neutron flux density, $n \cdot cm^{-2} \cdot s^{-1}$	$\sim 6.0 \cdot 10^{15}$
Driven fuel	Vi-pack-MOX, (PuN+UN)
Test fuel	Innovative fuels, MA fuels and targets
Core height, mm	600
Maximum core power density, kW/l	1100
Maximum neutron fluence per year, $n \cdot cm^{-2}$	$\sim 1 \cdot 10^{23}$ (up to 45 dpa)
Design lifetime, year	50
Number of autonomous test loops with different coolants	up to 4
Total number of experimental subassemblies and target devices for radioisotope production	up to 12 (core) up to 5 (radial shielding)
Number of experimental channels	up to 3 (core)
Number of experimental horizontal channels	up to 6 (outside reactor vessel)
Number of experimental vertical channels	up to 8 (outside reactor vessel)

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http://www.iaea.org/inisnkm/nkm/aws/fnss/index.html

Thanks for Your Attention !

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