The Abdus Salam
International Centre for Theoretical Physics

2141-3

Joint ICTP-IAEA Workshop on Nuclear Reaction Data for Advanced Reactor Technologies

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Fast Neutron Systems

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FAST REACTOR TECHNOLOGY DEVELOPMENT ACTIVITIES

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International Atomic Energy Agency

Outline

- Background
- Worldwide Fast Reactor Research and Technology Development Activities
  - China
  - France
  - India
  - Japan
  - Republic of Korea
  - European Union
  - Russia
  - USA
- Conclusions/Outlook

Worldwide Close to 400 FR-Years Cumulated Operation

- China
  - CEFR (23 MWe) 2010
- India
  - FBTR (13 MWe) 1985
  - PFBR (500 MWe) 2010/11
- Japan
  - Joyo (140 MWth) 1977
  - Monju (280 MWe) 1994
- Russia (USSR)
  - BR10 (8 MWth) 1958 – 2003
  - BOR60 (12 MWe) 1968
  - BN350 (130 MWe) 1972 – 99
  - BN600 (600 MWe) 1980
  - BN800 (870 MWe) 2012
- EU (D, F, UK)
  - Rapsodie (40 MWth) 1967 – 83
  - DFR (15 MWe) 1959 – 77
  - KNK-II (20 MWe) 1972 – 91
  - Phénix (250 MWe) 1973 – 2009
  - PFR (250 MWe) 1974 – 94
  - SNR300 (300 MWe) not started
  - Superphénix (1200 MWe) 1986 – 98
  - EFR Proj. (1580 MWe), cancelled 98
- USA
  - EBR-I (a few 100s We) 1951 – 64
  - EBR-II (20 MWe) 1961 – 1998
  - FFTF (400 MWth) 1980 – 1996
  - CRBR Proj.(380 MWe), cancelled 83

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Fast Reactors Today …

China, India, Russia

China's 25 MWe Experimental Fast Reactor (CEFR), Criticality Planned for 2010

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China's 25 MWe Experimental Fast Reactor (CEFR), Criticality Planned for 2010

CEFR, Outside View and Net

India's 500 MWe Prototype FBR (PFBR), Kalpakkam, Commissioning 2010-11

Safety Vessel (∅ 13.5 m, H 13.5 m,160 t) Transported from On-site Shop to Reactor Building (June 2008)

India's 500 MWe Prototype FBR (PFBR), Kalpakkam, Commissioning 2010-11

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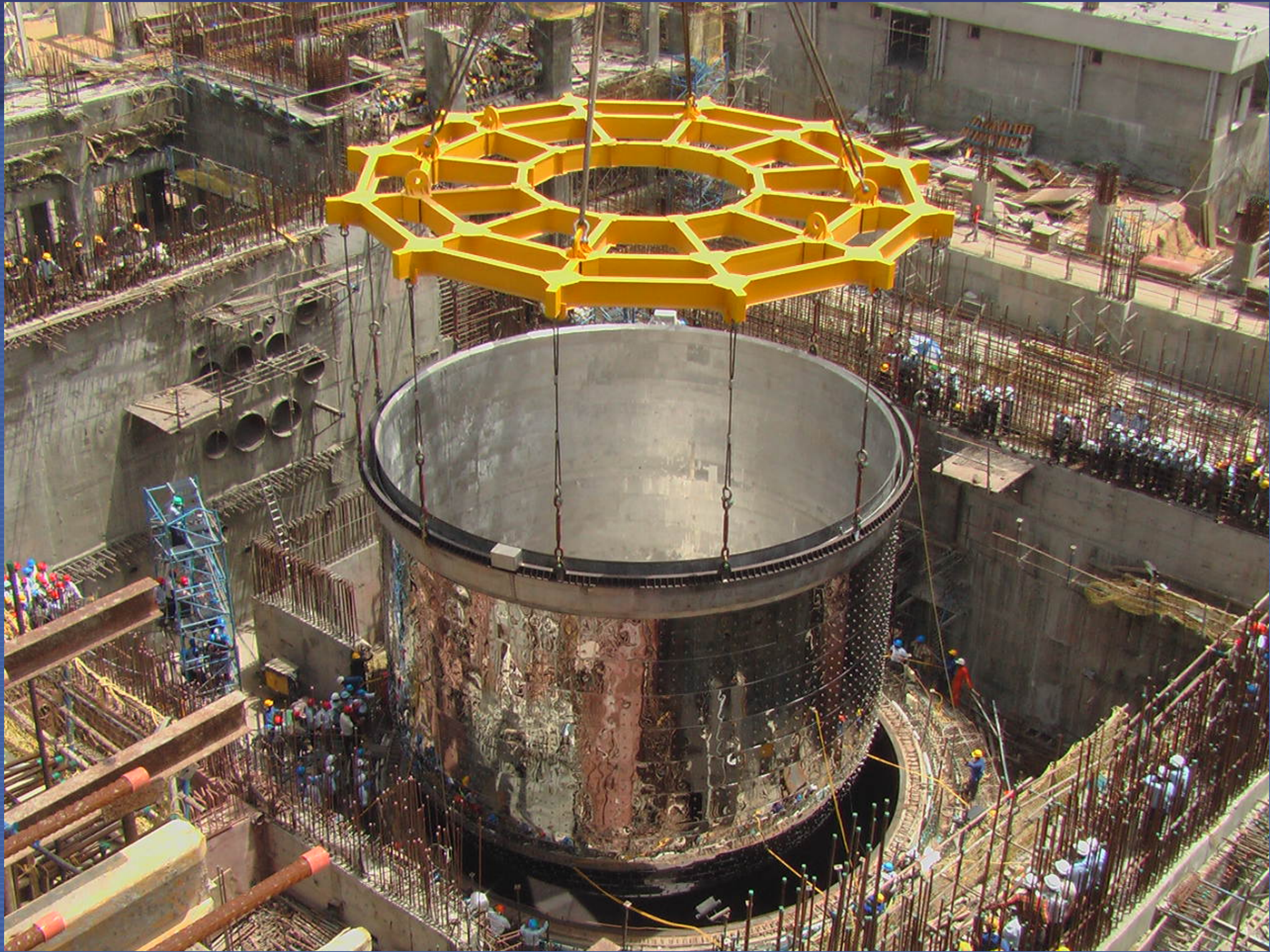
Safety Vessel Heaved Towards Reactor Vault (June 2008)

India's 500 MWe PFBR, Kalpakkam, Commissioning 2010-11



Safety Vessel Lowered into Reactor Vault (June 2008)

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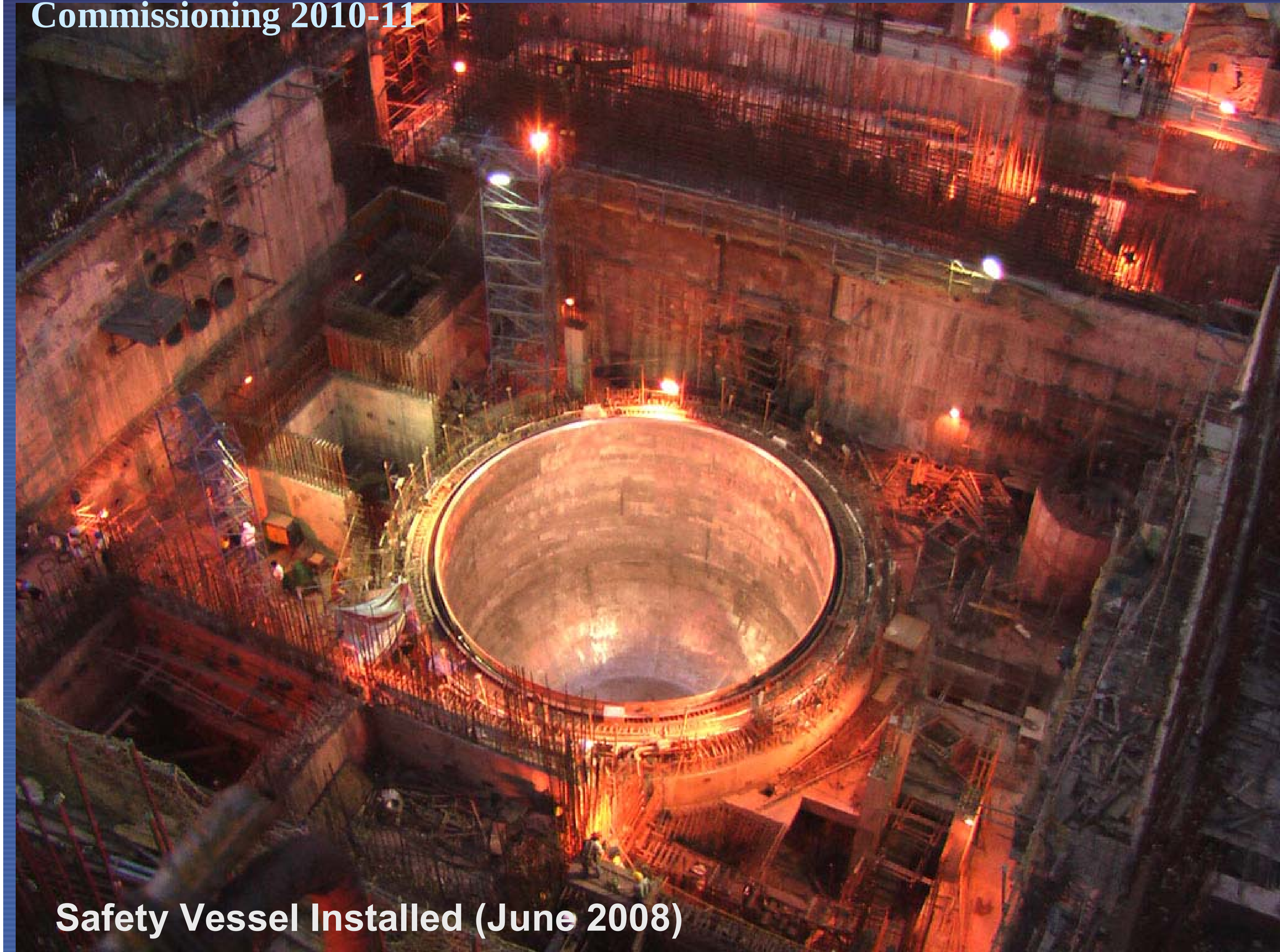
India's 500 MWe Prototype FBR (PFBR), Kalpakkam, Commissioning 2010-11



Safety Vessel Installed (June 2008)

India's 500 MWe Prototype FBR (PFBR), Kalpakkam, Commissioning 2010-11

Main Vessel Lowered into Safety Vessel (Dec. 2009)

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Russia's BN-800, Beloyarsk Site in September 2008 Commissioning Planned for 2013

Status of
Fast Reactor Research and
Technology Development Activities

CHINA

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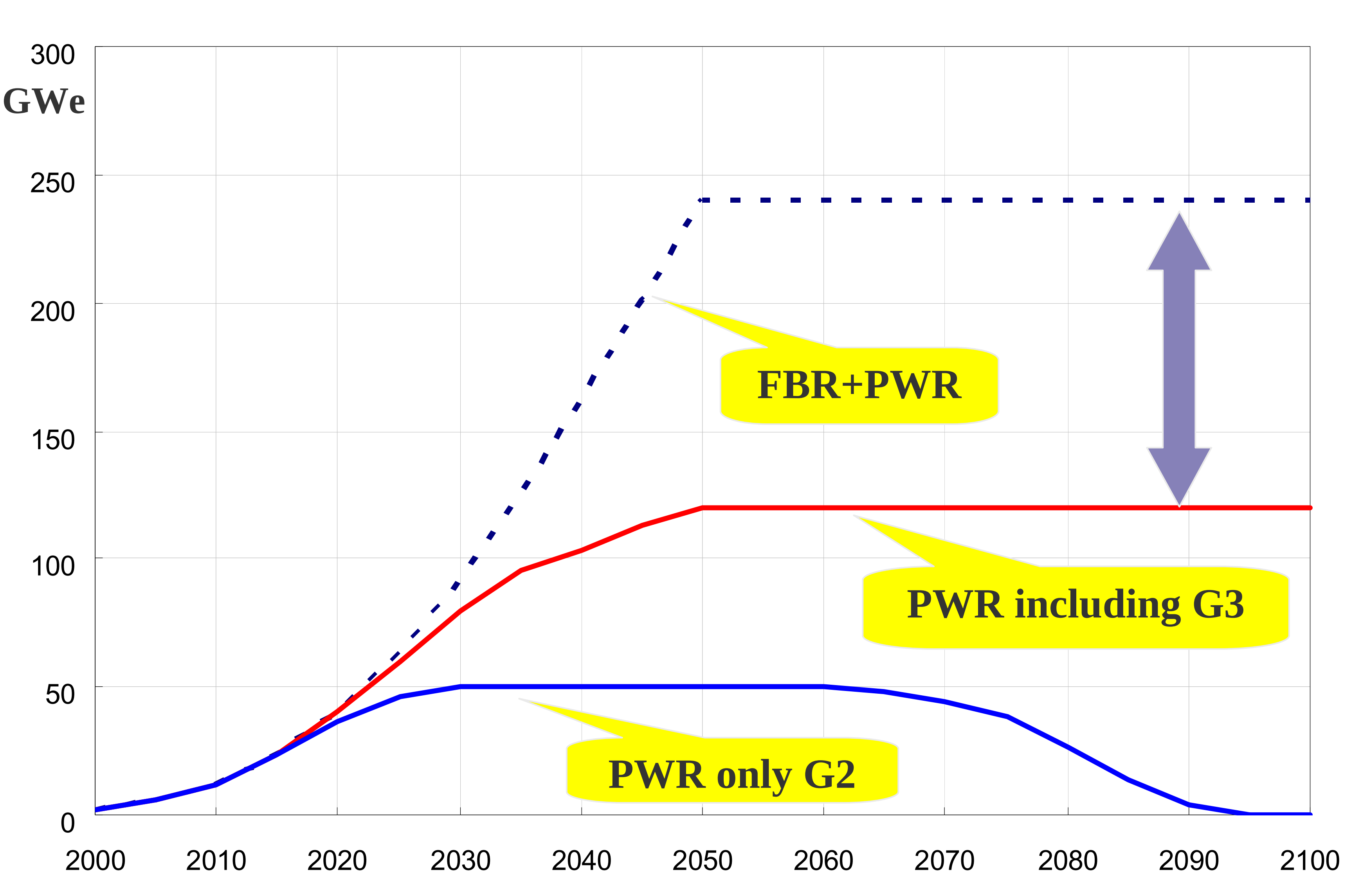
FRAMEWORK*

- China State Council approved “National Middle-long Term Science and Technology Development Program (2006-2020)”
  - Nuclear power strategic energy source
  - 18 GWe currently under construction, 2020 total nuclear power capacity goal 40 GWe
- Fast Reactor Technology Is One of the Program’s Four “Advanced Energy Technologies” Items
  - Defined objectives: completion of CEFR and mastering design, fuel and material technologies
  - No specific indications for the next steps, i.e. prototype fast reactor and associated fuel cycle facilities

***Based on a China Institute of Atomic Energy (CIAE) presentation at the IAEA Technical Meeting on “Design Features of Advanced Sodium Cooled Fast Reactors with Emphasis on Economics”, Vienna, 20 – 23 October 2008**

Optimistic Nuclear Energy Development Scenario



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CIAE Proposal/Vision for Sodium Cooled Fast Reactor (SFR) Development Program

First Step: CEFR (65MWth, 20MWe), 2010

Based on CEFR Progress/Experience
Establish SFR Development Roadmap

Second Step: CDFR (800-1000MWe), 2015-2020

Third Step: CCFR (>1000MWe), after 2025

CIAE Proposal/Vision for Sodium Cooled Fast Reactor (SFR) Development Program, cont'd

	CEFR	CDFR	CCFR
Power MWe	25	800	>1000
Coolant	Na	Na	Na
Type	Pool	Pool	Pool
Fuel	UO_2 (MOX)	MOX	MOX (Metal)
Cladding	Cr-Ni	Cr-Ni (ODS)	Cr-Ni (ODS)
Core Outlet Temp. °C	530	500-550	500-550
Linear Power W/cm	430	450-480	450-480
Burn-up MWd/kg_{HM}	60-100	100 (120)	120 (150)
Fuel Handling	DRPs, SMHM	DRPs, SMHM	DRPs, SMHM
Spent Fuel Storage	IVPS,WPSS	IVPS,WPSS	IVPS,WPSS
Safety	ASDS PDHRS	ASDS+PSDS PDHRS	ASDS+PSDS PDHRS

FRANCE

FRAMEWORK

- Law of 28 June 2006 on the sustainable management of nuclear materials and wastes requesting
  - The evaluation, by 2012, of the industrial feasibility of waste transmutation in GEN-IV systems, and
  - Its demonstration in a prototype system to be commissioned in 2020
- Objectives of the French Atomic Energy Committee announced on 20 Dec. 2006 with deadline 2012
  - Develop conceptual SFR designs with innovative options
  - Define specifications for SFR prototype aiming at validating innovative technology choices and recycle modes

SFR Experience

- MOX fuel performance in Phénix: 145 GWd/t, 16.8 % FIMA, 153 dpa (15-15 Ti cladding, EM10 wrapper)
- Feedback from large core operational experience (Superphénix)
- Validation and qualification of the ERANOS code system
- Safety tests with representative fuel elements (Cabri, Scarabee)
- Validation and qualification of pool an spray sodium fire simulation (Esmeralda)
- Structural mechanics: 10^5 hours Phénix operation, extensive feedback from Phénix lifetime extension program
- Plutonium multi-recycling (CAPRA)
- Utilization/transmutation of minor actinides and LLFPs (CADRA)

NEW GENERATION SFRs OBJECTIVES

❑French SFR Design Objectives ⟺ GEN-IV

- Optimisation of resources and non proliferation & physical protection ⇒ Break Even Core ⇒ Increased internal breeding gain
- Increased safety ⇒ prevention of severe accidents ⇒ reduction of ratio SVE/Doppler
- Waste management ⇒ Integral recycling of actinides, remote fabrication of TRU fuel

NEW GENERATION SFRs OBJECTIVES, cont'd

- **Sustainability**
  - **Efficient uranium utilization (closed fuel cycle)**
  - **Improve ultimate radioactive waste form (decay heat, radiotoxic inventory…)**
  - **Enhanced non-proliferation**
- **Robust safety demonstration matching at least Gen III safety criteria**
  - **Enhanced prevention of severe core damage**
  - **Practical exclusion of large energy release in hypothetical core meltdown**
  - **Contain the risk due to sodium's chemical reactivity**
  - **Robustness to external hazards**
- **Economic competitiveness**
  - **Reduction of capital cost and investment risk to about Gen III level**
  - **Improved plant operability (ISI&R, target of 90% availability, …)**
  - **Long lifetime (60 years)**

NEW GENERATION OF SFR

❑ASTRID (2020)

- 250 – 600 MWe
- Test-bed for innovation (systems, technologies)
- Capability for materials and fuel testing
- Advanced recycle modes demonstrations (U/Pu, MA)
- Improved U utilization
- Improved waste form

ASTRID

- ❑ Reduced excess reactivity
- ❑ Optimized reactivity effects (sodium void, Doppler, n-leakage…)
- ❑ Enhanced prevention/mitigation of severe accidents
  - **Design of subassembly (stiffness, flow, draining of molten fuel)**
  - **Natural convection as passive back-up core cooling system**
- ❑ **Improved core monitoring techniques**
- ❑ **Competitive generating costs**
  - **Flexible operating cycle (> 18 months)**
  - **Compact neutron shielding**
  - **High fuel burnup, extended lifetime of control rod absorbers**
- ❑ Potential for MA transmutation (homogeneous or heterogeneous recycling)

Preliminary proposal of an oxide fuel break-even, low reactivity swing core, with potential for breeding and MA transmutation, and ~20% gain on main safety criteria (void coefficient, core pressure drop)

ASTRID, cont'd

- **Improved reactivity coefficients**
  - Upper sodium plenum
  - In-core moderator
  - Innovative subassembly design
  - Large-diameter pins (~9.5 mm), small-diameter spacing wire (~1 mm)

  need for low swelling materials

  like F/M ODS, advanced austenitic steels

  CEA Reference: Ferritic Steel 14 – 18%Cr, Nano-structured ODS → Irradiations in Phénix (Super-nova, Matrix/Matrix 1)
  Future Development: Martensitic Steel 9%Cr, Nano-structured ODS)

ASTRID, cont'd

- **Innovative Fuels**
  - Carbide and nitride have increased margins to melting

Improve safety

Increase power density (Improved economy, HM Inventory)

ASTRID, CEA R&D OBJETIVES

- **2012 Milestone**
  - Safety assessment of optimized oxide core design with homogeneous and heterogeneous MA recycling
  - Assessment of the potential of advanced SFR fuels (carbide, metal, …)
  - Confirmation of ASTRID fuel development plan
  - Choice of materials for clad and wrapper, and associated design rules (RAMSES)

ASTRID ENERGY CONVERSION SYSTEMS

- ❑ Suppression or Limitation of Sodium / Water Interaction
- ❑ Cost Reduction → circuit simplification
- ❑ Improving thermal efficiency
- ❑ R&D
  - Gas conversion systems without intermediate Na loop
  - Compact intermediate loop with a fluid compatible with both sodium and water
  - Robust steam generators (double tubes, modular, …)

GAS CONVERSION SYSTEMS WITHOUT INTERMEDIATE SODIUM LOOP

- High Pressure Gas Brayton Cycle (N_2-He) Up to 39% Efficiency
- Supercritical CO_2 cycle Up to 43% Efficiency
- R&D
  - Gas entrainment risk
  - Na-CO_2 interaction
  - CO_2 carburization and oxidation
  - Technology of supercritical CO_2 cycle components
  - System behaviour of SFRs with supercritical CO_2 Brayton cycle power conversion systems
  - IHX

ASTRID REACTOR AND SUBSYSTEMS REVIEW

- ❑Pool / Loop Designs
- ❑Size Effects and Modular Designs
- ❑Assessment Criteria
  - Economics
  - Safety
  - ISI&R capability
  - Availability

ASTRID AND FLEXIBLE FUEL CYCLES

- **Advanced Fuel Cycles**
  - Resources
  - Waste minimization
  - Non-proliferation
- **Development of International Non-proliferation Standards Allowing Diverse Fuel Cycle Processes**
- **Keeping All Options Open for Possible Sequential Deployment**
  - U / Pu Recycling
  - Heterogeneous MA recycling (10 – 20% MA in blankets)
  - Homogeneous MA recycling (2 – 3% MA)

ASTRID AND FLEXIBLE FUEL CYCLES, cont'd

- Minor Actinides (MA) (Np, Am, Cm) Recycling in Blankets
  - Core 100 % $UPuO_2$
  - Blanket 100% $UMAO_2$, 10 – 40% MA/(U+MA)
- R&D
  - Recycling scenarios
  - Neutronics
  - Handling (heat generation, neutron sources)
  - Fabrication feasibility
  - Fuel pin and sub-assembly designs
  - Sub-assembly thermal hydraulics
  - Irradiation experiments

METHODS DEVELOPMENT

- ERANOS → Modular, Integrated, Deterministic Code System for Neutronics Analyses of Fast Reactor Cores
  - Various functions and calculation modules
  - Nuclear data libraries
  - “Formulaire” → Recommended design and reference calculation routes and associated biases and uncertainties, if possible
- R&D and Industrial Applications
- Study and Design of Conventional and Advanced SFRs
- Analysis of Physics Experiments in Zero-power Critical Facilities (MASURCA)
- Analysis of Irradiation Experiments in Power Reactors
- Power Reactor Core Performance Calculations, and Neutron and Gamma Shielding Calculations

METHODS DEVELOPMENT, cont'd

- Development of New Fast Reactor Core Design Software Suite
  - CONRAD → Nuclear data evaluation
  - GALILEE → Nuclear data treatment
  - DARWIN → Isotopic depletion
  - TRIPOLI → 3D multipurpose Monte-Carlo
  - APOLLO3 → 3D multipurpose deterministic

METHODS DEVELOPMENT, IMPROVEMENT OF CEA's "FORMULAIRE"

- 1997: Experimental Programmes (MASURCA, SNEAK, …) → Validation for "Classical" SFR Designs → ERANOS 1.2 / ERALIB1
- 2009: Validation of JEFF3.1 and of Calculation Routes / Monte-Carlo → ERANOS 2.1 / JEFF3.1 → Recommended Uncertainties and Calibrated Biases
- 2011: Model Improvements, Fine 3D Calculations, Sensitivities → ERANOS 2.2 / JEFF3.1 → Reduction of Method Bias
- 2013: Nuclear Data Improvement (Fe, Na, Actinides) and Covariance Matrices → ERANOS 2.3 / JEFF4 → Nuclear Data Uncertainty Reduction
- 2015: Experimental Programmes → GEN-IV Fast Reactor Validation → APOLLO3 / JEFF4

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Experimental Programmes

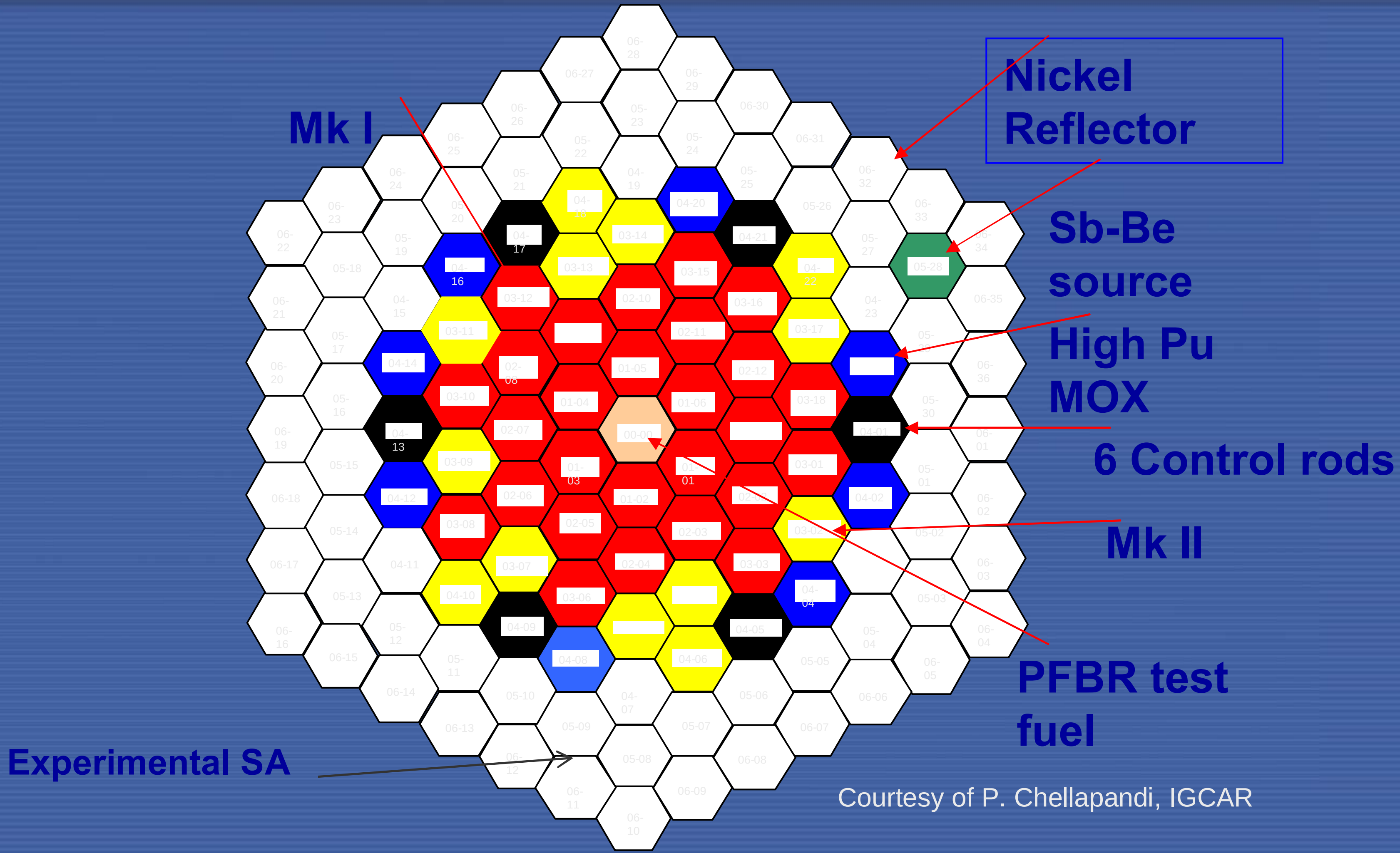
- Current Experimental Database is Insufficient for Validation of GEN-IV type SFRs, e.g.
  - Pu/HM too high → spectrum too hard, and adjoint flux (Φ^{+}) slope too low
  - Sodium volume fraction too high and no plenum
- CEA Considering New MASURCA Experimental Programmes: GENESIS and ENIGMA (for SFRs and GFRs, respectively)

INDIA

FAST BREEDER TEST REACTOR (FBTR)

- FBTR: Irradiation Facility For Development of Fuels and Core Materials
- Power: 40 MWth / 13.5 MWe
- Loop type
- Fuel: PuC (55 %) – UC (45 %)
- Linear Power Rate: 400 W/cm
- 15th Irradiation Campaign Since December 2008
- Carbide Fuel Burnup: 155 GWd/t
- PFBR (MOX) Fuel Burnup: 85.5 GWd/t
- Reactor Life Extension by further 20 years

FBTR 15th IRRADIATION CAMPAIGN



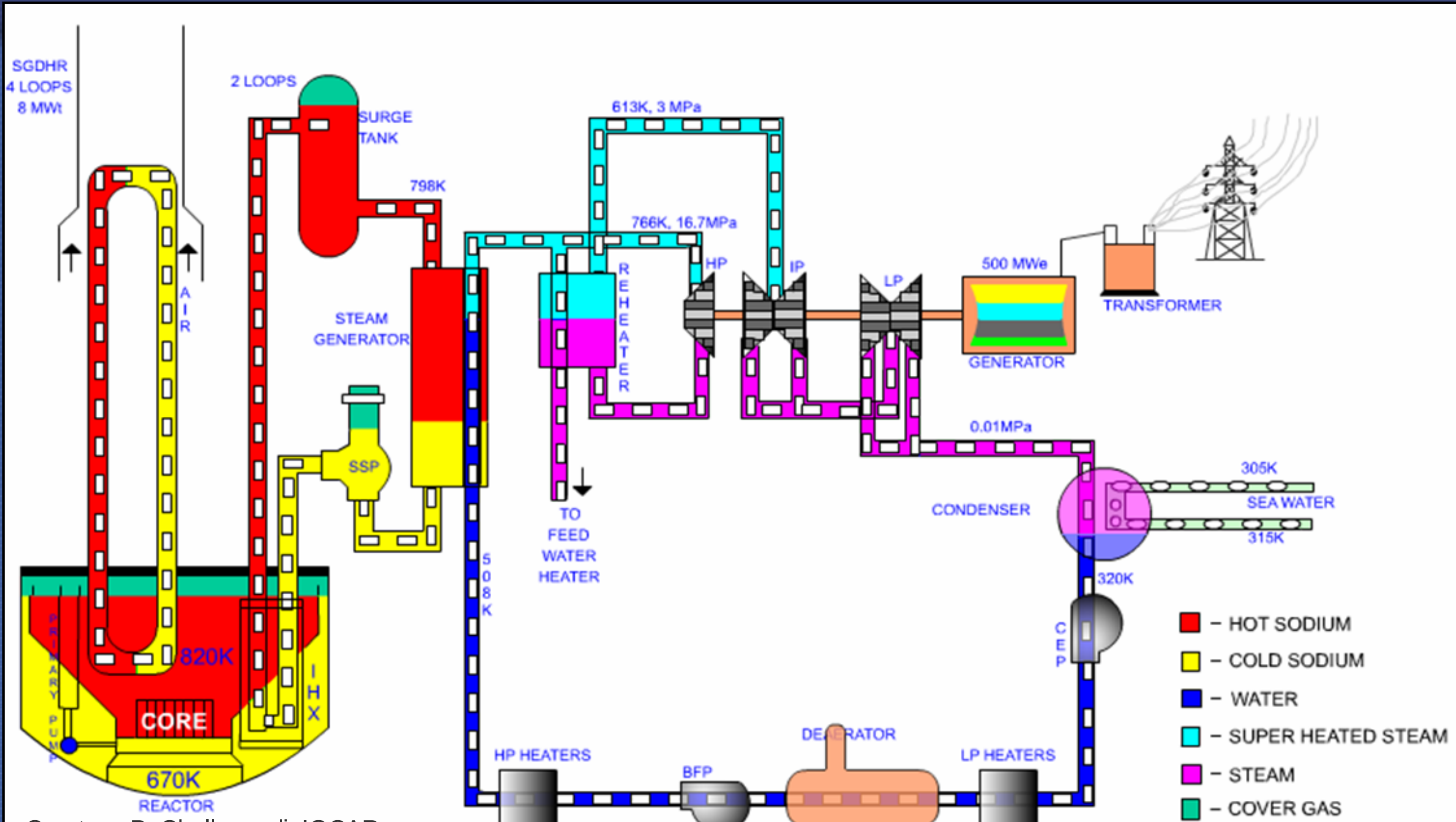
Courtesy of P. Chellapandi, IGCAR

Core (27 Mk I + 13 Mk II + 8 MOX + 1 PFBR Test Fuel + 1 SPA)

FBTR FUTURE PROGRAMME

- Seismic retro-fitting was completed in 2009
- Lead MK-I SA Discharged at End of 15^{th} Campaign (165 GWd/t) for PIE
- Yttria Capsule Discharged at End of 15^{th} Campaign for Isolating ^{90}Sr
- Continue Irradiation PFBR Test Fuel To Target burn-up (100 GWd/t) in 16^{th} Campaign
- From 17^{th} Campaign, It Is Planned to Irradiate Two Metallic Fuel Designs

PROTOTYPE FAST BREEDER REACTOR (PFBR)



Courtesy P. Chellapandi, IGCAR

Reactor power	- 1200 MWth / 500 MWe
Primary circuit concept	Pool
Fuel	UO_2-PuO_2
Steam condition	763 K at 16.7 MPa
Design Life	40 Year

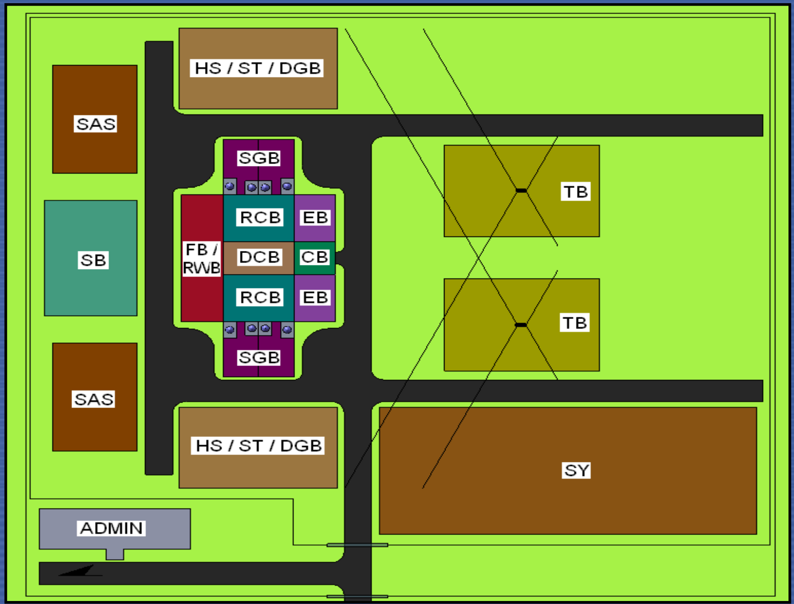
Number of PSP	2
Number of IHX	4
Number of sec loops	2
Number of SG per loop	4
Number of TG	1

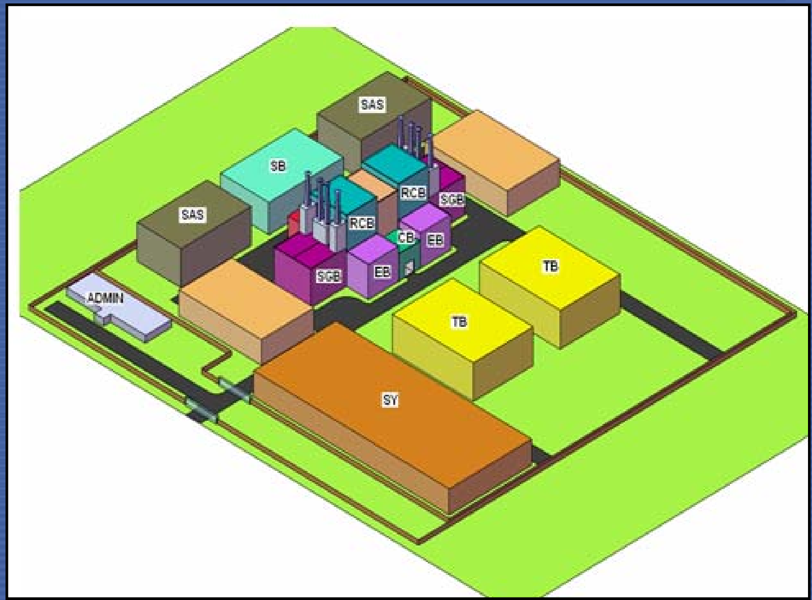
FUTURE COMMERCIAL FAST BREEDER REACTORS (CFBRs)

- **2013 – 2023: Six 500 MWe CFBRs Similar to PFBR With Improved Economics and Safety Characteristics**
  - MOX fuel
  - Twin unit concept
  - Two loop concept
  - Three SG modules per loop with increased length (30 m; PFBR has 4 modules per loop with 23 m length)
  - Optimum shielding
  - Use of 304 LN in place of 316 LN for cold pool components and piping
  - 85 % load factor
  - 60 years design life
  - Reduced construction time (5 y)
  - Enhanced burn up (up to 200 GWd/t to be achieved in stages)
- **One Twin Unit Sited at Kalpakkam adjacent to PFBR**
- **Site for Another Two Twin Units Under Consideration**
- **Features to Achieve CFBR Design Objectives, and Respective R&D Needs Under Discussion**
- **Beyond 2020: metallic fuelled, sodium cooled, 1000 MWe fast reactors**

CFBR TWIN UNIT

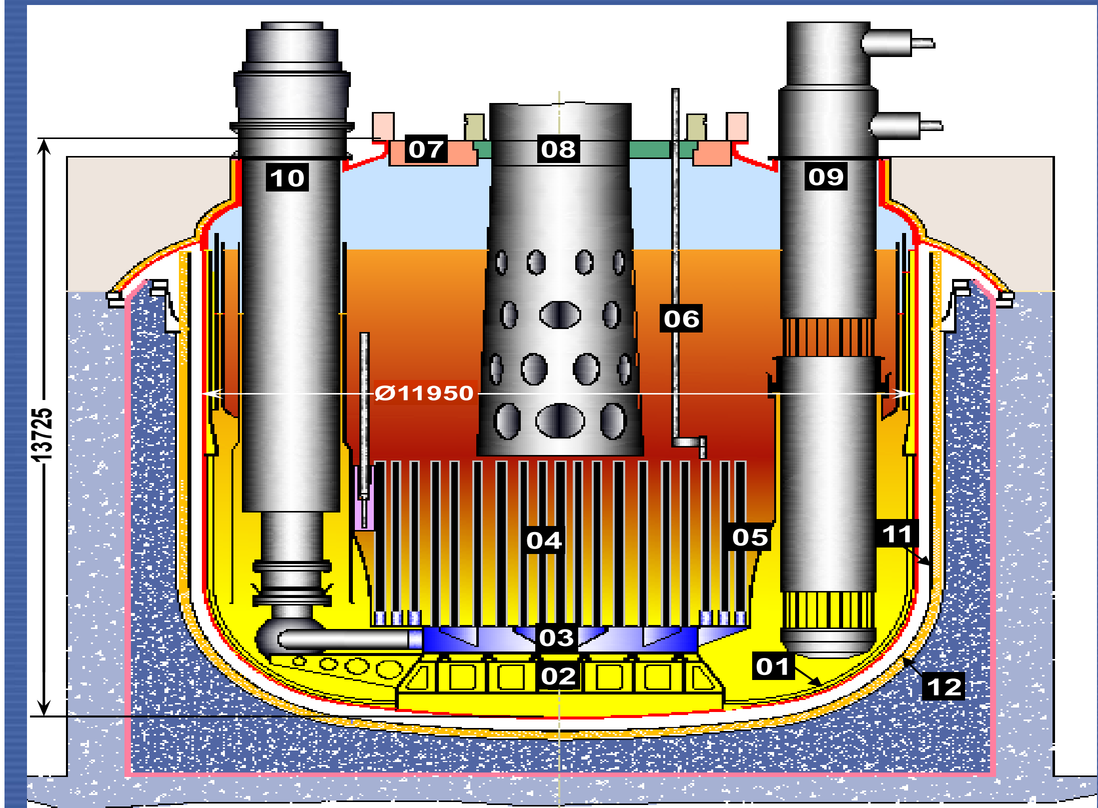
- Share Fuel & Rad-waste Building, Control Building, Turbine Building, Switchyard, Site Assembly Shop (Later Maintenance Building)
- Decontamination Building Common to Both Units Between Two Reactor Containment Buildings
- Compactness, Reduced Piping Length, etc…





Courtesy P. Chellapandi, IGCAR

CFBR



- Innovation under consideration
  - Compact and symmetric welded grid plate without fuel transfer post
  - Inner vessel having single curved redan integrated with fuel transfer post
  - Thick plate rotatable plugs
  - Control plug integrated with small rotatable plug
  - Torus shaped thick plate roof slab
  - Simplified fuel handling scheme with elimination of inclined fuel transfer machine
  - Torus support skirt for reactor assembly with optimum support location to minimize the seismic moments
  - Safety vessel made of carbon steel integrated with reactor vault liner
- Detailed thermal hydraulics and structural mechanics analyses are under progress to arrive at optimum dimensions and structural wall thicknesses in compliance with the design code RCC-MR (2007)

R&D ACTIVITIES (IGCAR)

- Reactor Physics
  - IGCAR-CEA Collaboration on Fast Reactor Safety
  - End of Life Neutronics Tests in PHENIX
  - Core physics studies for metallic fuels with zirconium
- Thermal Hydraulics Studies of DHR System
- Seismic Qualification Tests
- Diverse Safety Rod Drop Time Measurements Using Acoustic and Ultrasonic Techniques
- Component Development
  - Diverse Safety Rod Drive Mechanism
  - High temperature ultrasonic transducer for scanning under sodium

R&D ACTIVITIES (IGCAR), cont'd

- Material Studies
  - Optimisation studies for D9 clad material
  - Fe-0.11C-9Cr-2W-0.2Ti-0.35Y2O3 Oxide dispersion strengthened (ODS) alloy developed employing mechanical milling and hot extrusion
  - Studies of the creep-fatigue behaviour of Mod.9Cr-1Mo steel base metal and weld joint in air at 873K
  - Characterization of the influence of simulated service exposure (643-823K) on fracture toughness and fatigue crack growth behaviour of Indian SS 316N welds
- Safety Studies
- Fuel Cycle

JAPAN

Main Reference: JAEA Country Report at the
42nd TWG-FR Annual Meeting, May 2009, Kalpakkam, India

Framework

- Oct 05 – Nuclear Energy Policy (Atomic Energy Commission of Japan AEC)
  - Promote R&D for introduction of Commercial FR Cycle Technology
  - Feasibility Study to establish the FR cycle technologies by ~2015
  - Commercial introduction of FR cycle ~2050
- Mar 06 – Science and Technology Basic Plan (Council for Science and Technology Policy)
  - FR cycle technology one of the key technologies having national importance
- Jul/Aug 06 – Report on Nuclear Energy Policy [Ministry of Education, Culture, Sports, Science and Technology (MEXT), and Ministry of Economy, Trade and Industry (METI)]
  - Joint MEXT, METI, JAEA, utilities & vendors council to study FR cycle technology demonstration
  - Development of a demonstration FR and introduction ~2025
- Dec 06 – Basic Policy on FR Cycle Technologies R&D Over Next Decade (AEC)
  - MEXT, METI and JAEA, in collaboration with utilities, vendors and universities coordinate R&D on selected concept
  - JAEA implement FR Cycle Technology Commercialization R&D program; deliver conceptual designs for both demonstration and commercial facilities by 2015
  - Monju restart (2009?) with two objectives to be attained within 10 years: gain operational experience and R&D facility for the FR Cycle Commercialization Program

Fast Reactor Cycle Technology Development Project (FaCT)

2005	2010	~2015
FS	*FaCT Project*	

FS Phase II

Obtained results

- Selection of promising concepts & principles for R&D prioritization
- R&D program until ~2015

FaCT Phase I

Expected results

- Determination of innovative technologies
- Establishment of commercial reactor and fuel cycle concepts

FaCT Phase II

Expected results

- Definition of main data based on experimental and simulation studies
- Definition of specification for commercial facilities
- Definition of technology demonstration program leading to FR Cycle commercialization

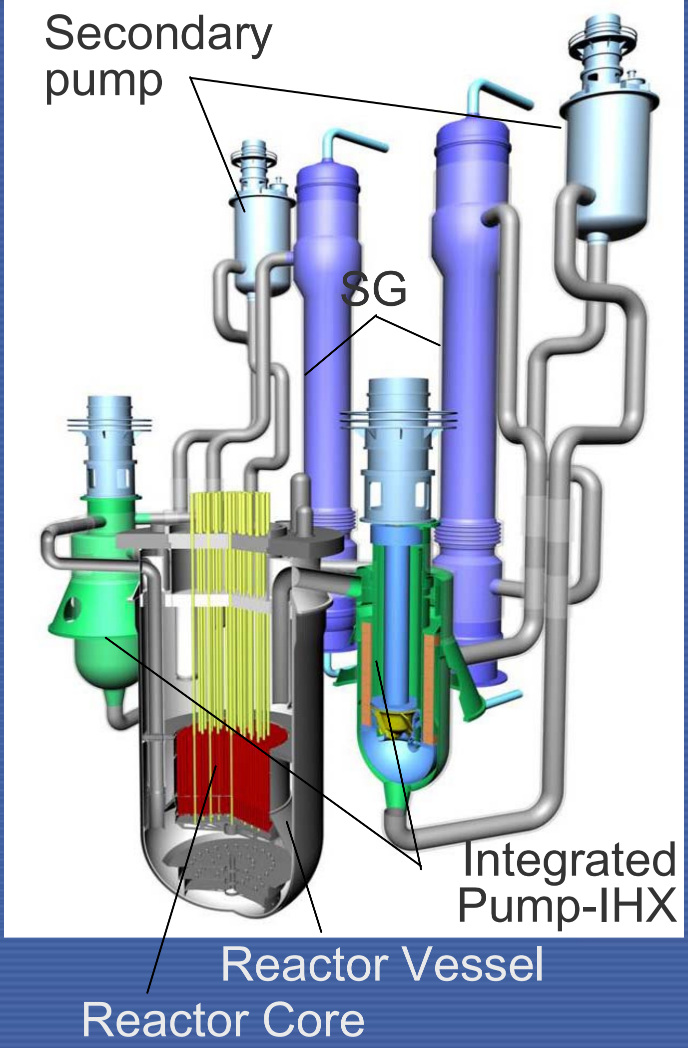
Courtesy H. Negishi, JAEA

FaCT Project Development Objectives

- Safety and Reliability → At Least Equal to Future LWRs
- Sustainability
  - Environment protection → reduction of waste volume and radiotoxity
  - Waste management → improvement of fuel cycle backend
  - Efficient utilization of fuel resources → breeding/"burning" flexibility
- Economic Competitiveness → Reduction of Construction, Operation and Fuel Cycle Costs
- Non Proliferation → Safeguards and Physical Protection According to FR Cycle Features

JAEA SODIUM COOLED FAST REACTOR (JSFR)

Parameter	Specifications
Power	3570 MWt / 1500 MWe
Number of loops	2
Primary sodium outlet temperature and flow rate	550 / 395 °C 3.24×10^7 kg/h/loop
Secondary sodium temperature and flow rate	520 / 335 ° C 2.70×10^7 kg/h/loop
Main steam temperature and pressure	497 °C 19.2 MPa
Feed water temperature and flow rate	240 °C 5.77×10^6 kg/h
Plant efficiency	~42%
Fuel type	TRU-MOX
Breeding ratio	Break even (1.03), 1.1, 1.2
Cycle length	26 months or less, 4 batches



Courtesy H. Negishi, JAEA

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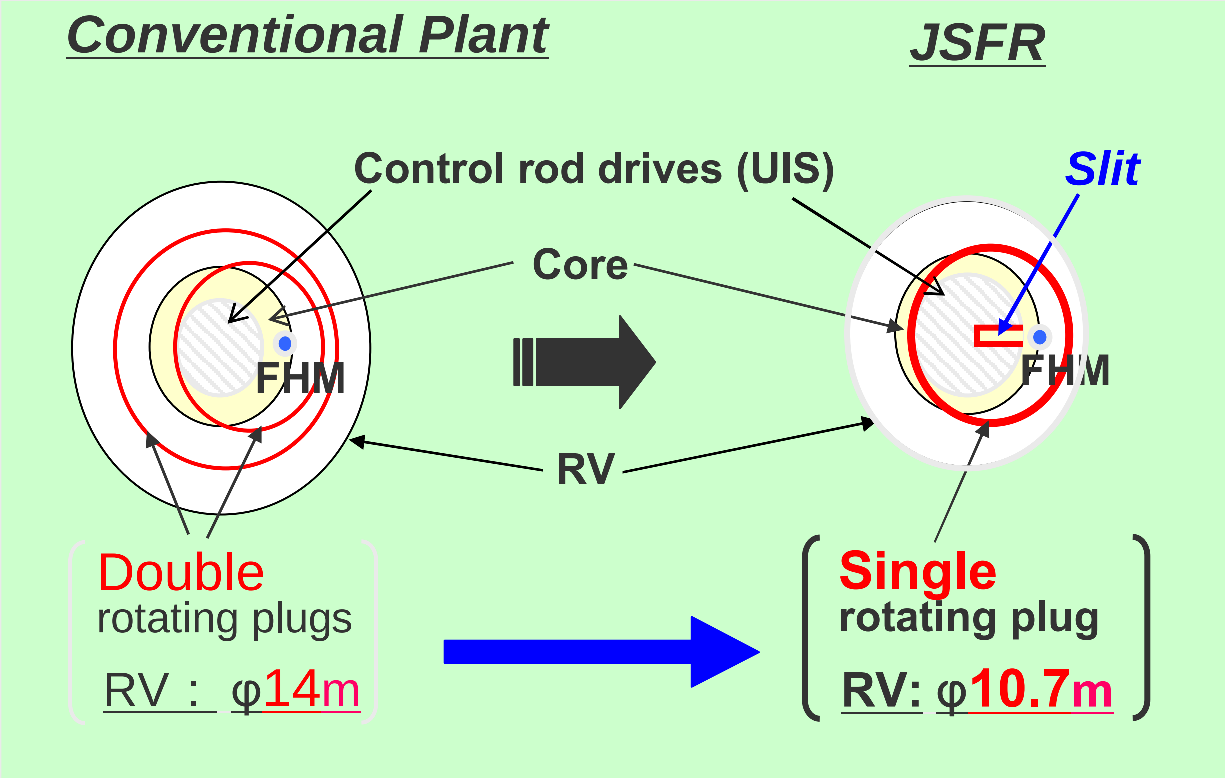
JSFR COST REDUCTION MEASURES

- Compact Reactor Vessel
  - Fuel handling machine suitable for the slit in the reduced upper internal structure (UIS) area
  - Zr-H Shielding
  - Suppression of wall cooling layers
- Simplified Reactor Internal Structure
- Heat Transport System with Shortened Piping
- Two Loop Cooling System
- Compact Design of the Reactor Building
- Operation Cost Reduction

JSFR COMPACT REACTOR VESSEL

New fuel handling system without large double rotating plugs enables compact reactor vessel



Compared to double rotating plugs

Courtesy H. Negishi, JAEA

- Reactor vessel diameter 3.3m smaller
- Reactor structure material amount reduced by ~40%

JSFR OPERATION COST REDUCTION

High performance core (high internal conversion ratio)

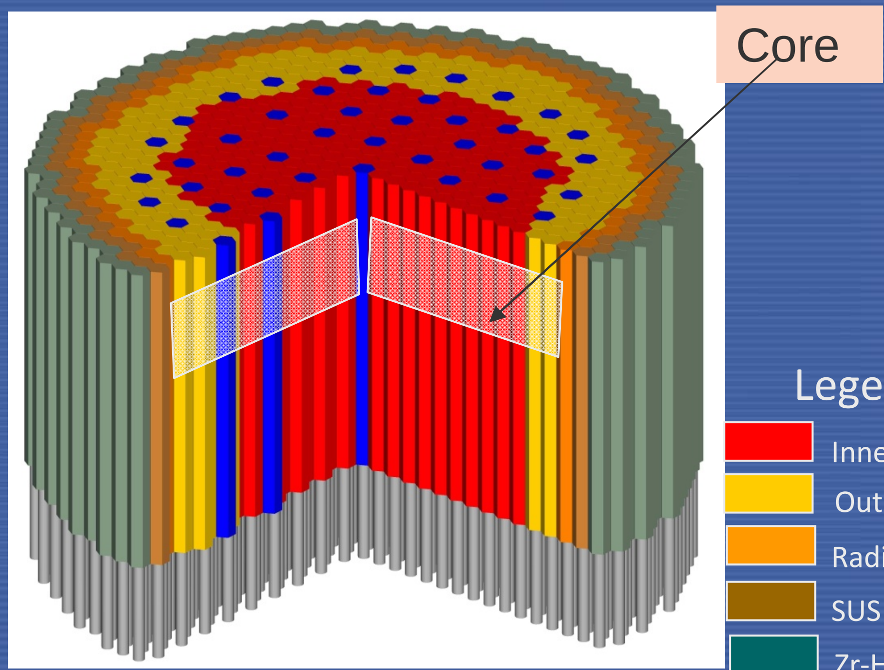
- ***Operation cycle length: 26 months***
- ***Discharge burn up: 150 GWd/t (core average)***
  90 GWd/t (core + blanket average)

Reducing Operation Cost

Main Neutronic Characteristics

Output power (MWth)	3570
Cycle length (months)	26
Refueling batch [core/RB]	4/4
Pu-enrichment (wt%) [Inner/outer]	18.3/20.9
Burnup reactivity (%dk/kk')	2.3
Breeding ratio	1.10
Discharge burnup (GWd/t) [core]	147
[core + blanket]	90
Pu fissile inventory (t/GWe)	5.7
Maximum neutron fluence (n/cm^2)	5E23
Sodium void reactivity ($)	5.3

Courtesy H. Negishi, JAEA



Legend
- Inner core
- Outer core
- Radial blanket
- SUS shield
- Zr-H shield
- Control rod

Reactor core cut view

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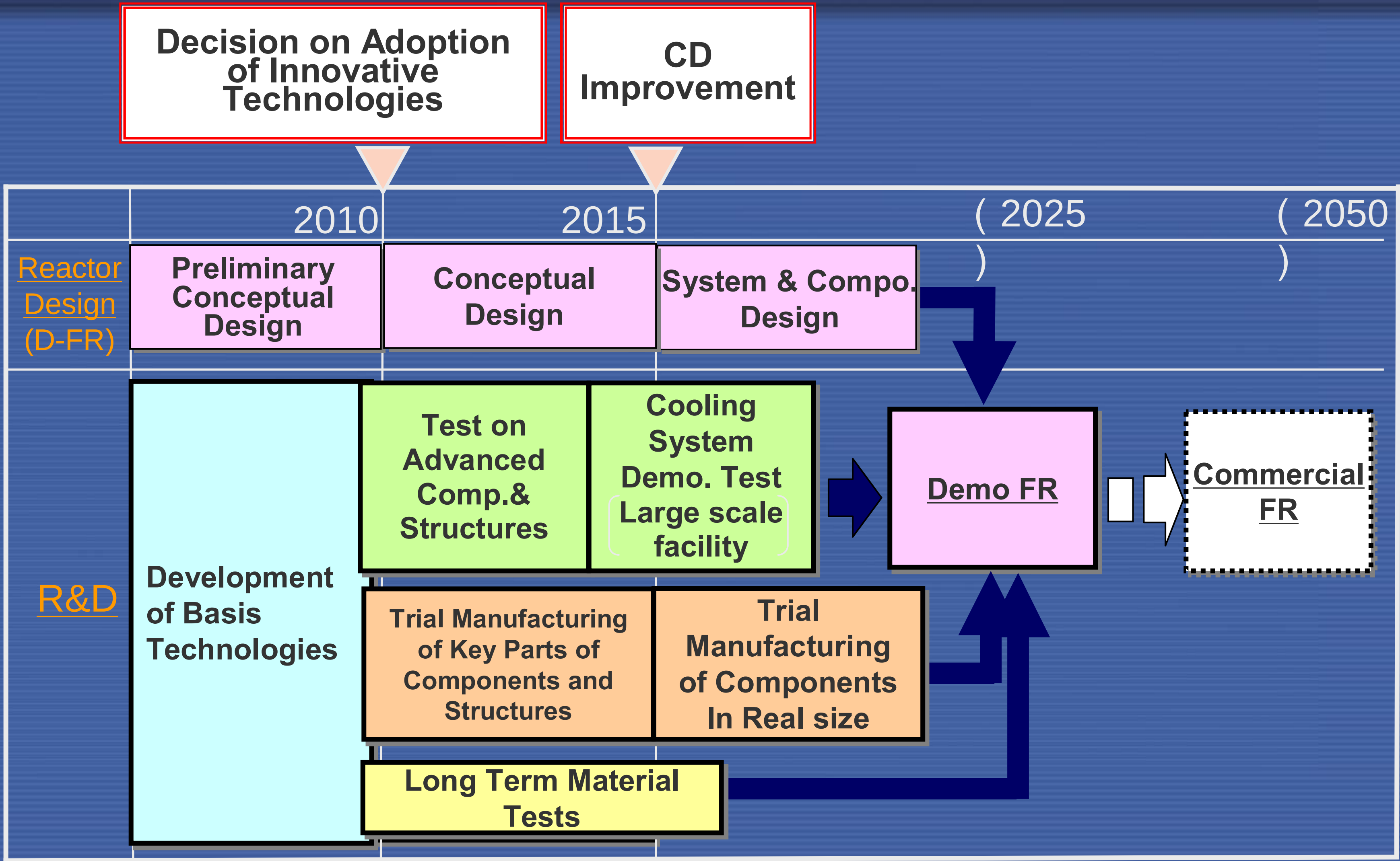
JSFR MAJOR RESEARCH AND TECHNOLOGY DEVELOPMENT AREAS

- **Two-loop Cooling System**
  - Hydraulic and structural integrity due increased coolant flow rate per loop
  - Safety issues: piping break/failure, reliability of decay heat removal system
- **Reactor Structure**
  - Inspection capability
  - Seismic design margin
- **Simplified Fuel Handling System**
  - Precise positioning with rapid movement of fuel handling machine
  - Short outage time for refuelling
  - Dry cleaning of spent fuel to reduce the amount of liquid rad-waste
  - Efficient and safe TRU bearing fuel handling
- **Passive Reactor Shutdown System**
  - Self Actuated Shutdown System (SASS) as backup shutdown system
- **MA Containing Oxide Fuel Core**
  - Flexibility during LWR-to-FR cycle transition stage

JSFR DEVELOPMENT ROADMAP

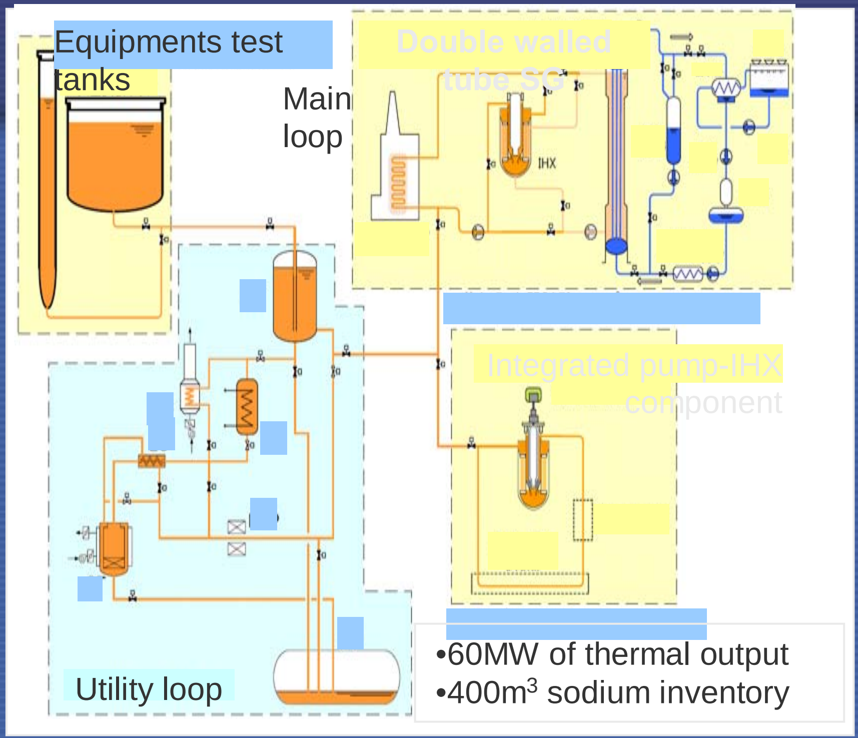


Courtesy H. Negishi, JAEA

PLANNED JAEA LARGE SCALE SODIUM TEST FACILITY

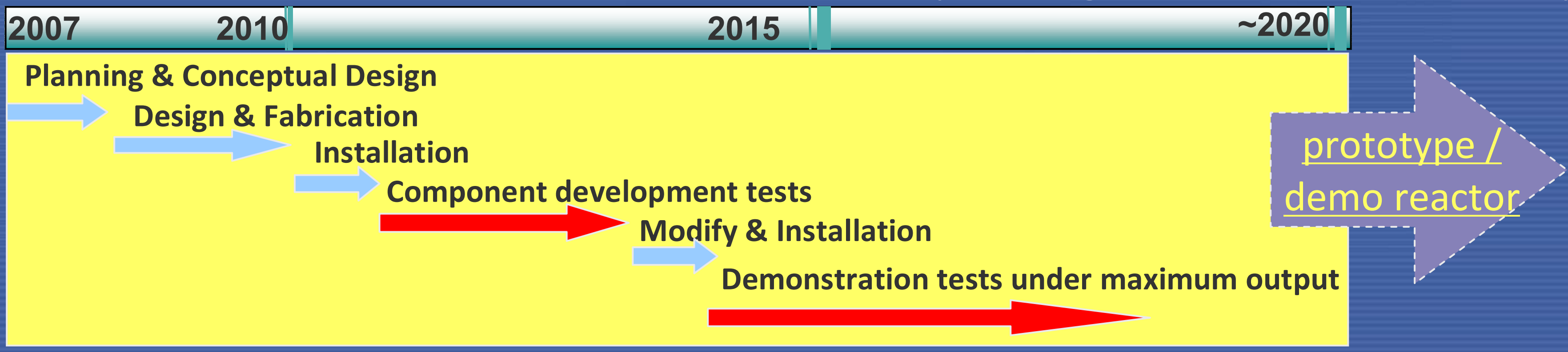
Objectives

- ❑Component development tests
  - Double walled tube SG
  - Integrated pump-IHX
- ❑System and components demonstration tests
  - Primary loop, IHX, secondary loop, SG, and steam-water system
- ❑Equipment demonstration in sodium



Concept of Large-Scale Sodium Test Facility

Schedule :



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Courtesy H. Negishi, JAEA

REPUBLIC OF KOREA

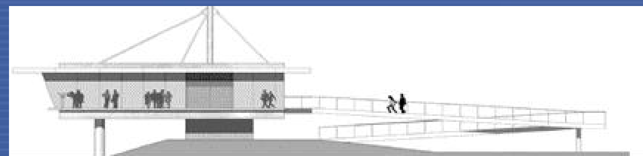
FRAMEWORK

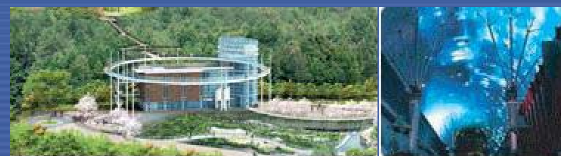
- 26 Feb 2008: Radioactive Waste Management (RWM) Law For Safe Management of Radioactive Wastes Including Spent Fuels
  - 1 Jan 2009: Establishment of Korea Radioactive Waste Management Corporation (KRMC)
  - Establishment of RWM fund for low/intermediate level radioactive waste and for spent fuel
  - Establishment of a basic plan for RWM with the approval of the Korea Atomic Energy Commission (KAEC)

CONSTRUCTION OF YOLSONG NUCLEAR ENVIRONMNET CENTRE

- Low and Intermediate Level Rad-Waste Disposal Repository
  - Completion of construction by June 2010
  - Final geological disposal
  - Size of 1st stage 10^5 drums (final stage: 8×10^5 drums)

1. Main Building
2. Observatory Platform
3. Entrance Disposal Cave
4. Outlook





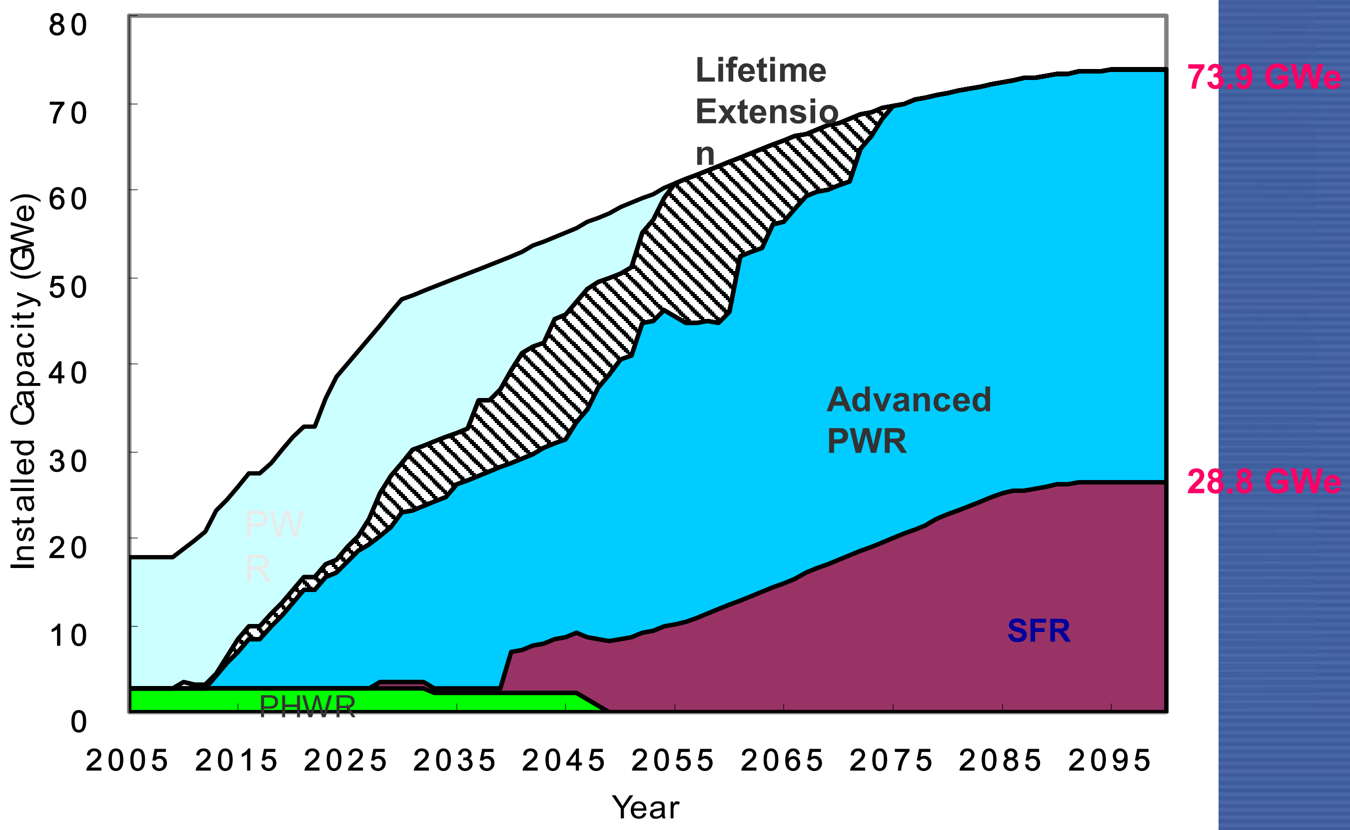
Courtesy HAHN Dohee and CHANG Jinwook , KAERI

GREEN GROWTH INITIATIVE

- 13 Jan 2009: National Science and Technology Committee Selected Green Technologies for R&D Support
  - Five areas, 27 technologies ⇨ "Energy Technology Area" ⇨ "Environment Friendly, Proliferation Resistant Sodium Fast Reactor and Pyro-Process System Technology"

TRANSITION SCENARIO (KAERI)

- ❑ Total electricity generation growth rate
  - 2006 – 2030 : Planned
  - 2031 – 2050 : 1.0%/year
  - 2051 – 2100 : Reduced to 0%/year in 2100
- ❑ Nuclear share of 59.0% after 2030



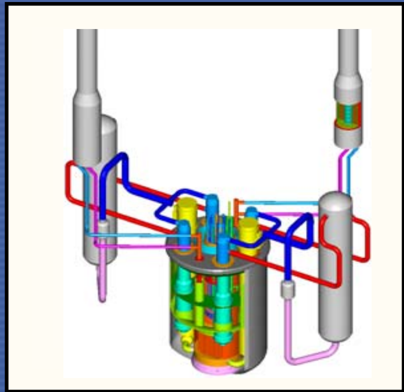
Courtesy HAHN Dohee and CHANG Jinwook , KAERI

STATUS OF SFR TECHNOLOGY DEVELOPMENT

’07

GEN-IV SFR Development

’02



KALIMER-600 Conceptual Design

’97

KALIMER-150 Conceptual Design

’92

Basic Research

Courtesy HAHN Dohee and CHANG Jinwook , KAERI

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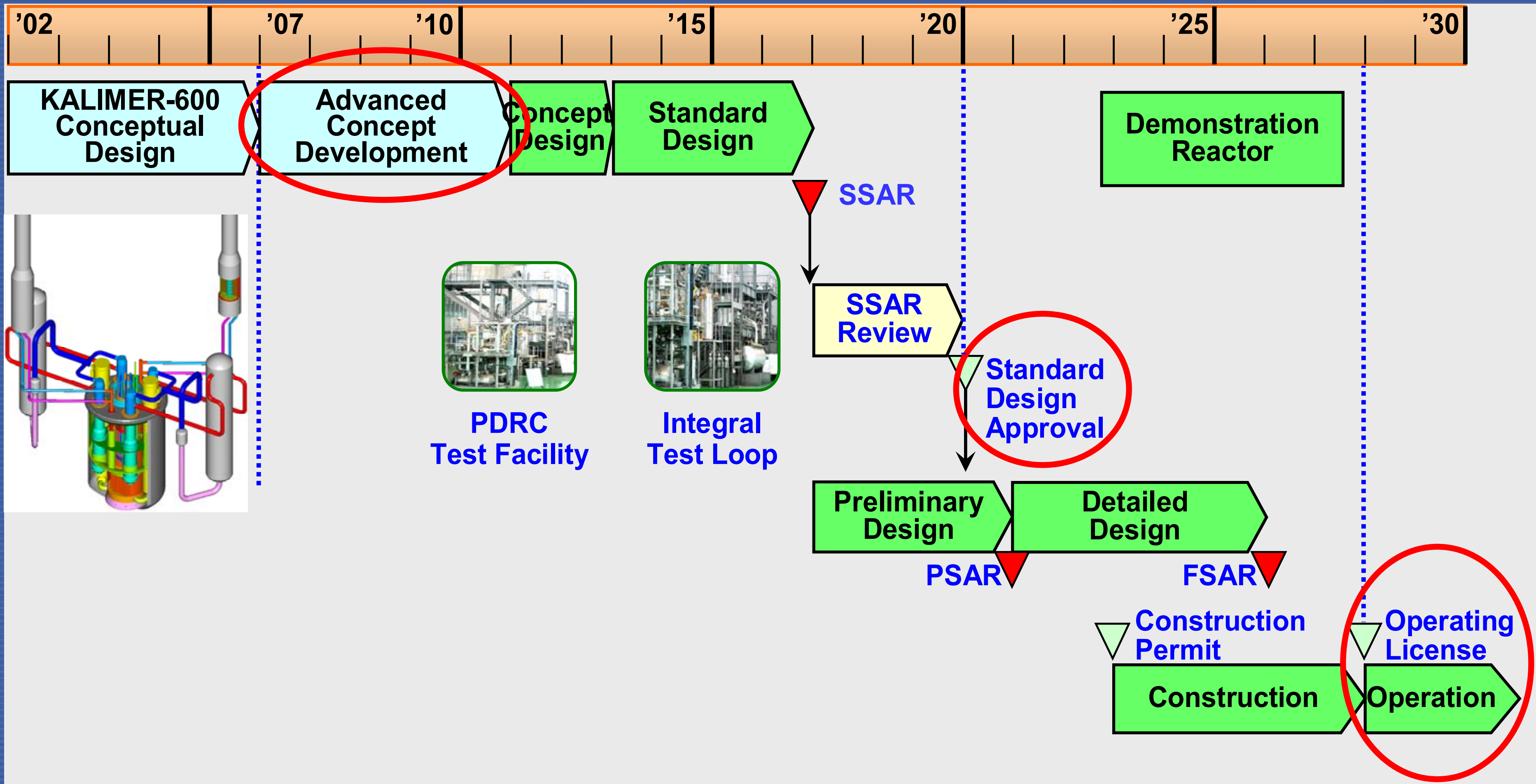
LONG-TERM DEVELOPMENT

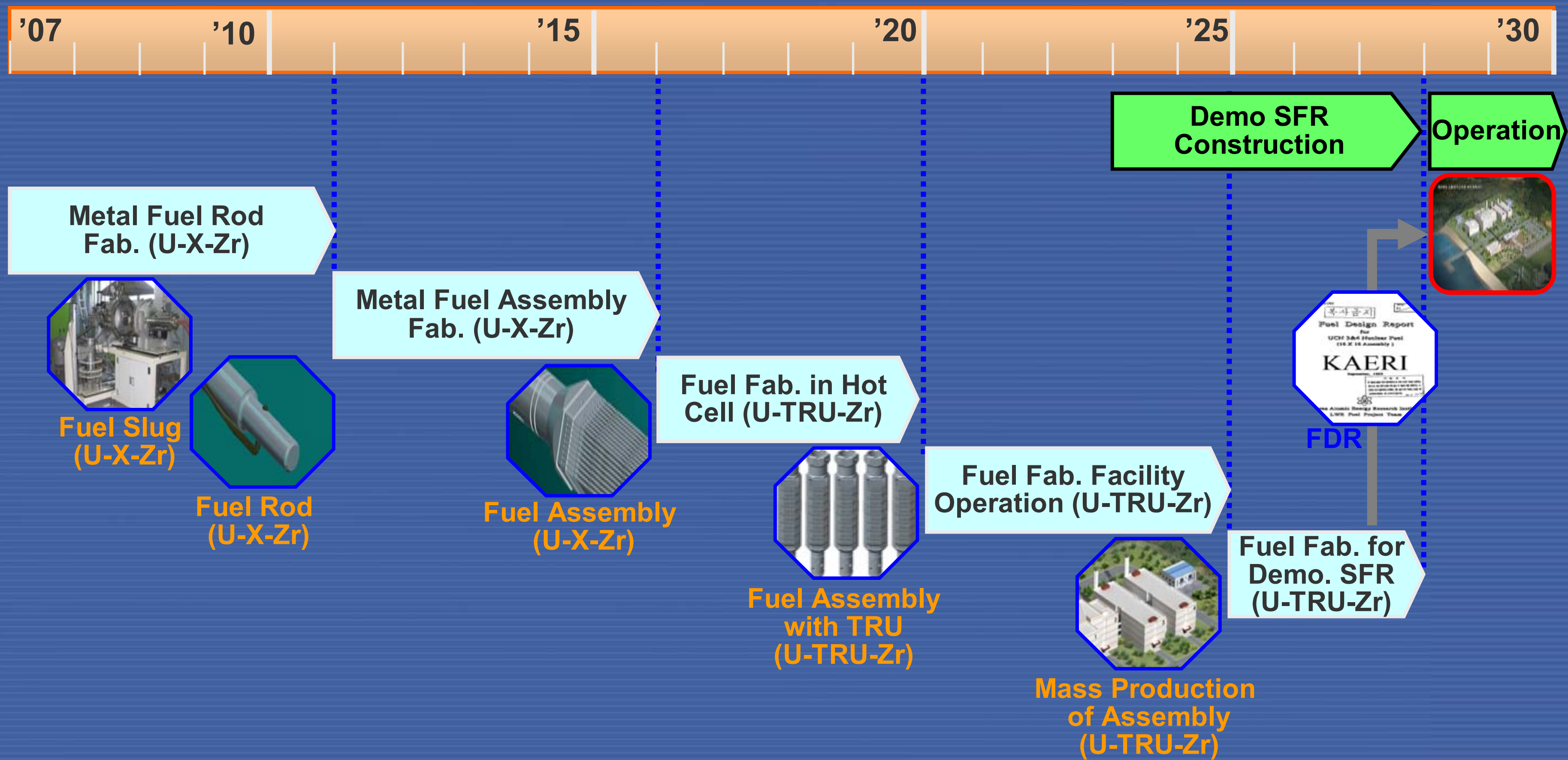
- 22 Dec 2008: KAEC Approved Long-term Development Plans for Future Reactor Systems
  - Provide consistent roadmap for long-term R&D activities
  - Addresses SFR, Pyroprocess, and VHTR
- Detailed Implementation Plan Under Discussion, Defining Schedule, Deliverables, Responsibilities and Resources
- Implementation through R&D Programs of Korean Science and Engineering Foundation (KOSEF) With Funds From Korean Ministry of Education, Science and Technology (MEST)

SFR LONG-TERM DEVELOPMENT PLAN



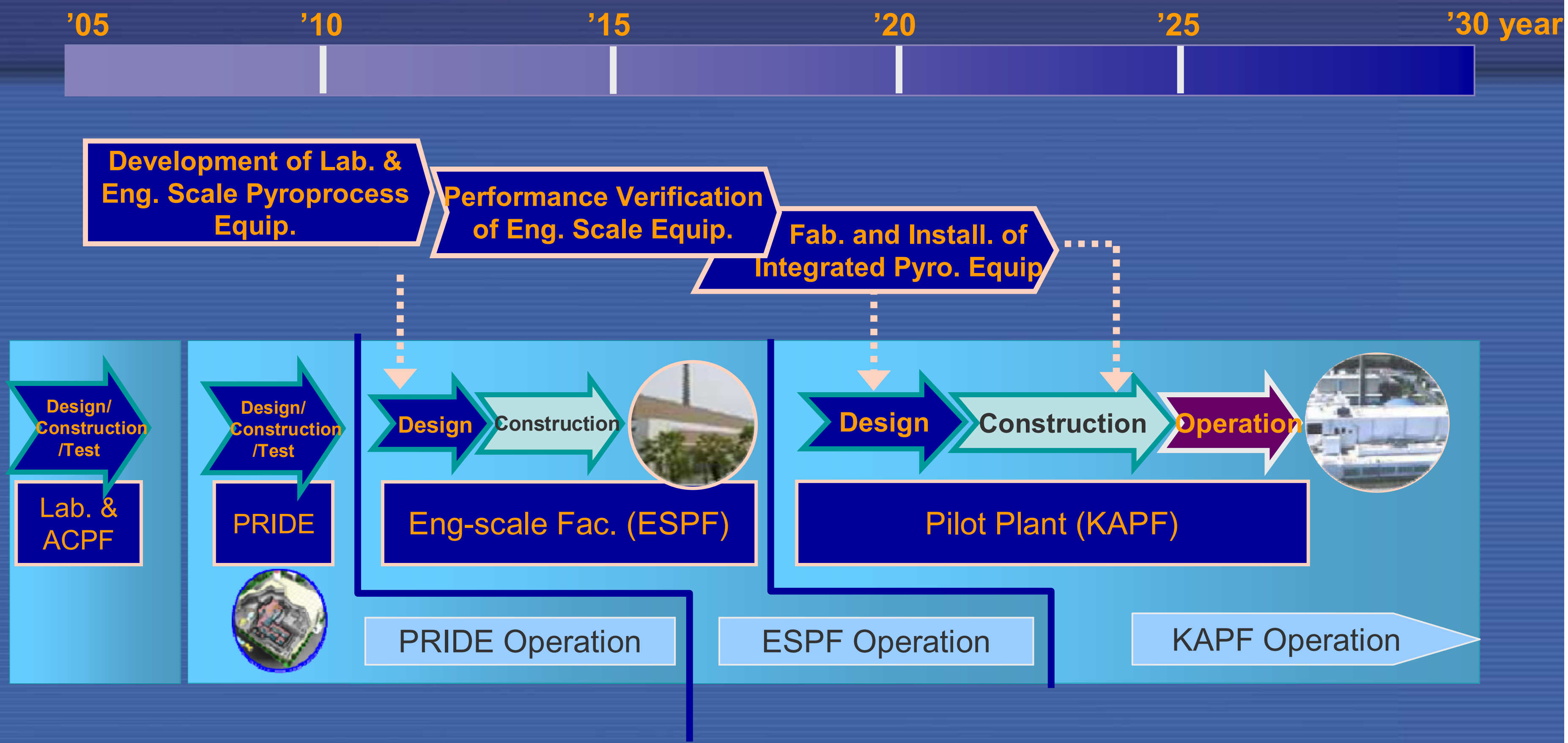
Courtesy HAHN Dohee and CHANG Jinwook , KAERI

METAL FUEL LONG-TERM DEVELOPMENT PLAN



Courtesy HAHN Dohee and CHANG Jinwook , KAERI

PYROPROCESS LONG-TERM DEVELOPMENT PLAN



Legend

- **ACPF: Advanced spent fuel Conditioning Process Facility**
- **PRIDE: PyRoprocess Integrated inactive DEmonstration facility**
- **ESPF: Engineering-Scale Pyroprocess Facility**
- **KAPF: Korea Advanced Pyroprocess Facility**

Courtesy HAHN Dohee and CHANG Jinwook , KAERI

ADVACNED SFR DESIGN

- 1200 MWe, Pool-type
- U-TRU-Zr Fuel
- Core I/O Temp 390 / 545 □
- Passive Decay Heat Removal Circuit System (PDRC)
- 2-Loop Intermediate Heat Transport System/Steam Generation System
- Net Efficiency 39.4%

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PASSIVE DECAY HEAT REMOVAL CIRCUIT

❑ Design Features

- Elimination of active components
- Operation by natural circulation
- No operator action

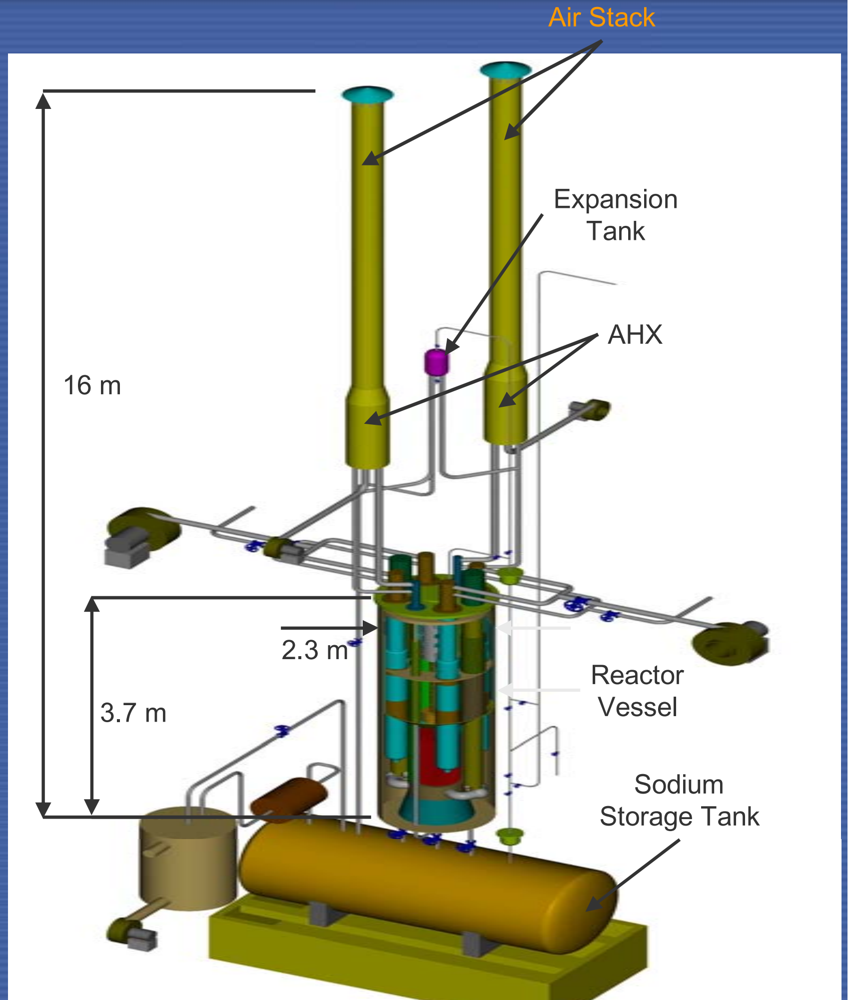
❑ System Operation

- Normal operation ⇒ Minimized heat loss to prevent sodium freezing
- Primary pump trip ⇒ Decay heat removal by natural circulation

PASSIVE DECAY HEAT REMOVAL CIRCUIT EXPERIMENT

- Verification of design concept
- Confirmation of basic design issues
- Assessment of initial & long-term cooling capability by natural circulation
- Verification of heat removal capability by transient mode
- Dynamic simulation of natural circulation cool-down during key design basis events
- Prevention of sodium solidification
- Countermeasures for a postulated RV fracture
- Establishment of database for validating system analysis code

Preliminary Concept of Main Test Section



AHX: Na-Air Heat Exchanger

Courtesy HAHN Dohee and CHANG Jinwook , KAERI

KAERI R&D OBJECTIVES AND ACTIVITIES

Technology Goals of GEN-IV Reactor System

Sustainability	Proliferation Resistance	Economics	Safety
▪ Burner CR: 0.5-0.8 ▪ Breakeven CR: 1.0	▪ No Blanket	▪ FOAK - 4 ¢/kWh - 2,000 $/kWe	▪ CDF < 10^{-6}/R·Y ▪ Grace time of more than three days

R & D

- Burner core development
- Breakeven core development
- Development of MA bearing metal fuel

- Size optimization
- Core and structure optimization
- $S\text{-}CO_2$ Brayton cycle system
- Integrated components
- High temp LBB

PDRC concept validation
⇒ Passive safety

Validation of analysis tools
⇒ Inherent safety

Conceptual Design of Advanced SFR

- Core concept without blankets
- Construction cost ~ $2,300/kWe
- PDRC Concept
- Safety analysis tools

IAEA Courtesy HAHN Dohee and CHANG Jinwook , KAERI

European Union

FRAMEWORK

- Sustainable Nuclear Energy Technology Platform (SNE-TP) ⇒ Broad European R&D Stakeholder Initiative In Nuclear Safety and Systems Launched in September 2007
  - R&D priority to industrial applications
  - Large experimental facilities needed
  - "Joint Undertakings" as prototype realization frameworks
  - March 2009 – "Strategic Research Agenda (SRA)"
- Europe's Low Carbon Energy Technology Policy ⇒ Strategic Energy Technology (SET) Plan
  - Includes nuclear fission and fusion
  - Includes the "European Industrial Initiative (EII) on Sustainable Fission" which is well aligned with the SRA of the SNE-TP

SRA of the SNE-TP FOR FAST REACTORS

- GEN-IV Fast Reactor and Closed Fuel Cycle ⇒ SFR, LFR, GFR, ADS
  - Innovative fuels and materials
  - Safety rules
  - R&D infrastructure
  - Simulation and experiments

EUROPEAN FAST REACTOR STRATEGY

- Reference Concept ⇒ SFR
- Alternative Concepts ⇒ LFR and GFR
- Supporting Infrastructure ⇒ Facilities for Research, Irradiation, and Fuel Fabrication
- Goal at 2020 Horizon ⇒
  - SFR Prototype (250 – 600 MWe)
  - LFR or GFR Demo (50 – 100 MWth)
  - MOX Fuel Fabrication Unit
  - MA “Micro-pilot” Fuel Fabrication Unit
- Goal at 2040 Horizon ⇒ Deployment of GEN-IV Fast Reactors (Earlier If New Energy Needs)

RELATED EURATOM FP7 PROJECTS

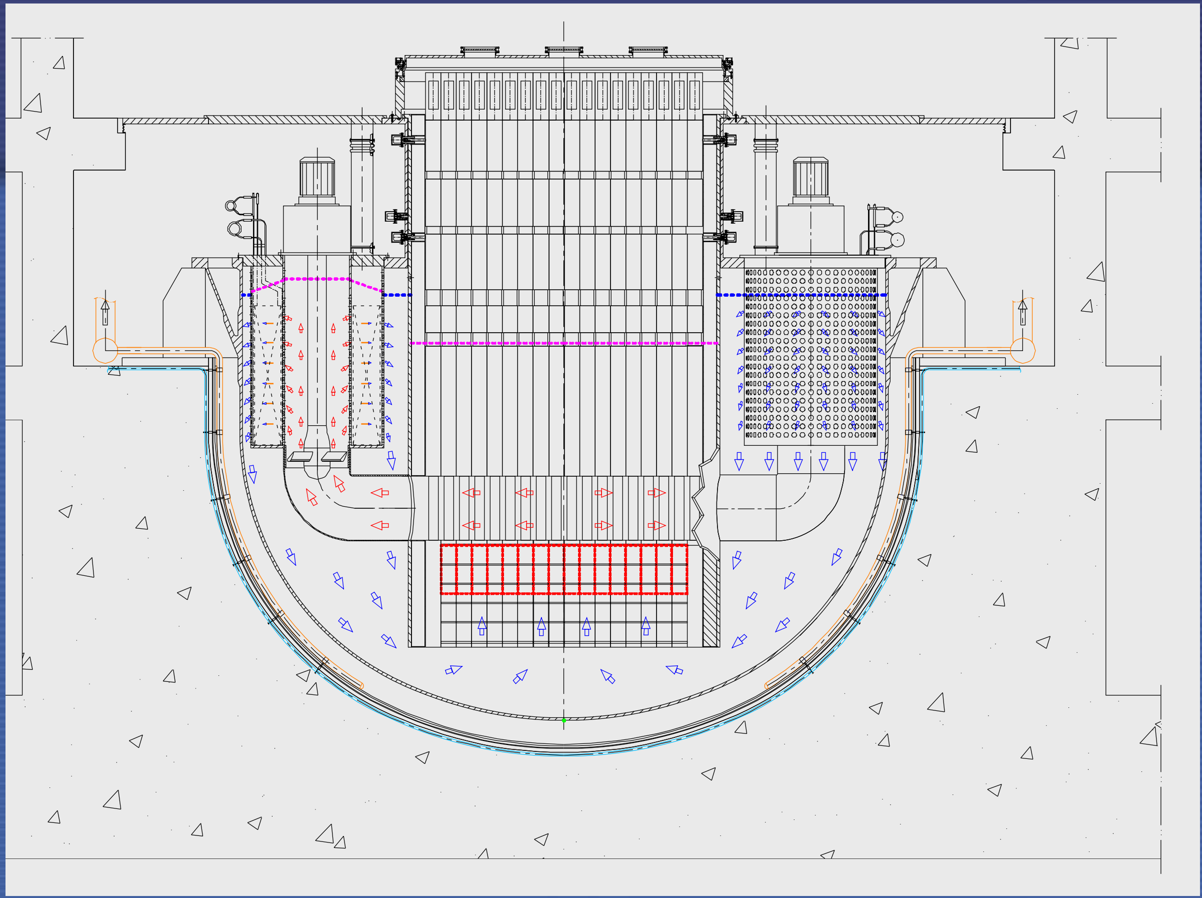
- ESFR (European Sodium Cooled Fast Reactor collaborative project)
- LEADER (Lead Cooled Fast Reactor collaborative project)
- GoFastR (European Gas Cooled Fast Reactor collaborative project)
- CDT (Central Design Team) ➡ MYRRHA/FASTEF (Fast Transmutation Experimental Facility)

ESFR

- R&D to Substantiate Key Viability and Performance Issues
- Support Development of a GEN-IV European SFR (European SFR Prototype ASTRID)
- EU Contributions to GEN-IV International Forum (GIF)

LEADER

- Builds on FP6 Project ELSY (European Lead Cooled System)
- Conceptual Design of 50 – 100 MWth ETPP (European Technology Pilot Plant)



ELSY Concept: 600 MWe, compact primary circuit, no intermediate cooling system, secondary water loop with Rankine cycle, open square pitch fuel assemblies

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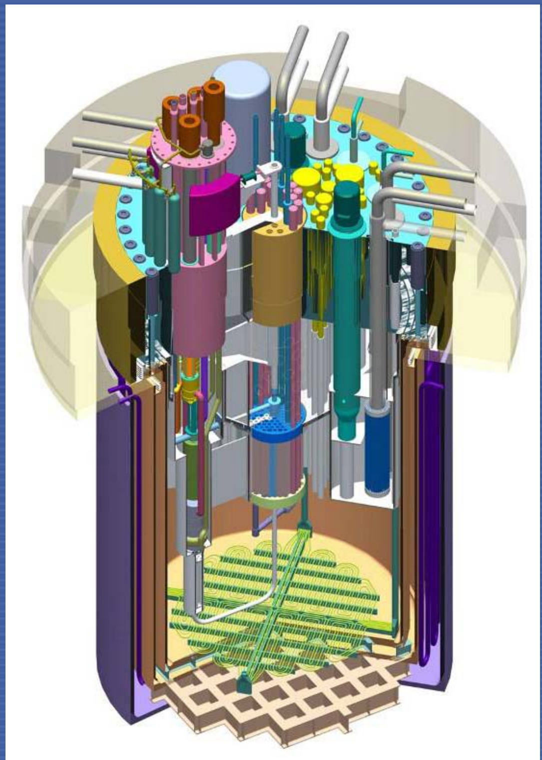
GoFastR

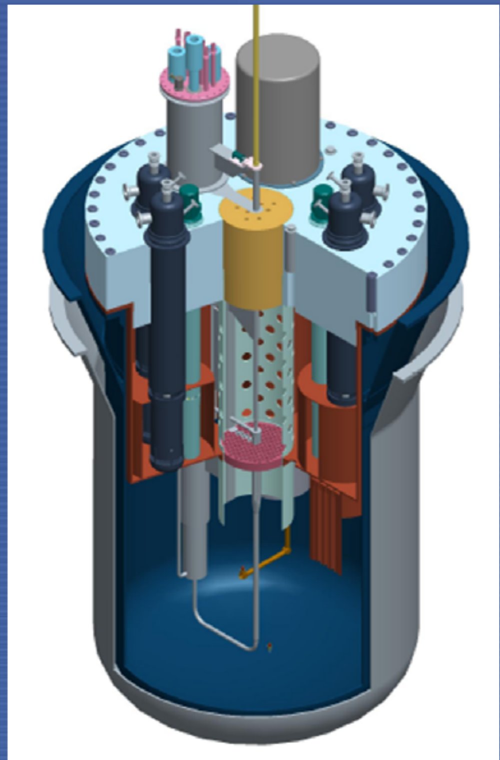
- Builds on FP5 and FP6 Projects (GCFR)
- Objective: Develop a "Sustainable" VHTR
  - Outlet temperature ~850 °C
  - Compact core ~100 MW_{th}/m^3 power density
  - Low Pu inventory, self sustaining
- Design Challenges
  - Fuel development
  - Loss of flow / loss of coolant (depressurization)

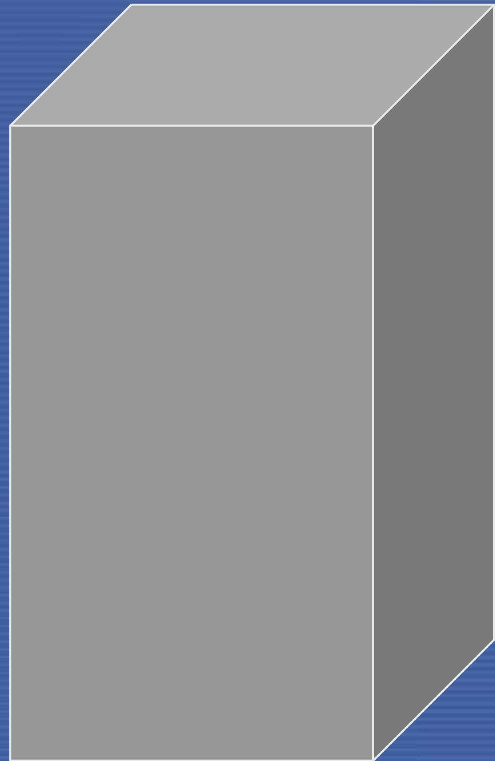
CDT, MYRRHA/FASTEF

- Build on FP5 and FP6 projects (MYRRHA/XT-ADS)
- FASTEF
  - Flexible fast spectrum irradiation facility
  - A full step ADS demo facility and P&T testing facility
  - Contribute to the demonstration of heavy liquid metal technology LFR
- CDT (Central Design Team) Project
  - Set up of a centralised multi-disciplinary team at SCK•CEN in Mol
  - Produce advanced design of a flexible fast spectrum irradiation facility operating in sub-critical mode (ADS) and critical
  - Create the nucleus of the “Owner Engineering Team” for the realization of MYRRHA/FASTEF

CDT, MYRRHA/FASTEF, cont'd







MYRRHA	⇨	XT-ADS	⇨	FASTEF
2005		2009		2012

Courtesy D. De Bruyn, SCK•CEN

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Russian Federation

Federal Target Program (FTP) For Nuclear Power Technology of a New Generation for the Period 2010-2020

- Objectives
  - Enhancing safety
  - Closing the fuel cycle
- Mid-term plan
  - Fast reactor technology without construction of new light water reactors
  - Continue operation of existing light water reactors with recycle of spent fuel in the next generation fast reactors
- Fast reactor program based on extensive operational experience with experimental and industrial sodium cooled fast reactors
- Development and experience gained with heavy liquid metal cooled (Pb and Pb-Bi eutectic alloy) fast reactor technology
- Sodium cooled, mixed uranium-plutonium oxide fuelled BN-800 under construction, commissioning in 2013

Federal Target Program (FTP) For Nuclear Power Technology of a New Generation for the Period 2010-2020

- The fast reactor development program includes
  - Life extension: experimental reactor BOR-60 and industrial reactor BN-600 (the latter has ended in April 2010)
  - Design of new (BOR-60 replacement) experimental reactor MBIR [100 MWth/50 MWel, sodium cooled, uranium-plutonium oxide (alternatively uranium-plutonium nitride) fuelled]
  - Simultaneous development of sodium, lead, & lead-bismuth eutectic alloy cooled fast reactors (SFR, BREST-OD-300, SVBR-100, respectively) and of their respective fuel cycles
  - Design of advanced large-size sodium cooled commercial fast reactor BN-K

USA

FAST REACTOR PROGRAM

- SFR Preferred Option
- Objective: Develop and Demonstrate Advanced Fast Recycle Reactor With Closed Fuel Cycle
- R&D Program
  - Closed fuel cycle demonstration
  - Experimental infrastructure
  - Capital cost reduction
  - Safety validation
  - Advanced reactor simulation

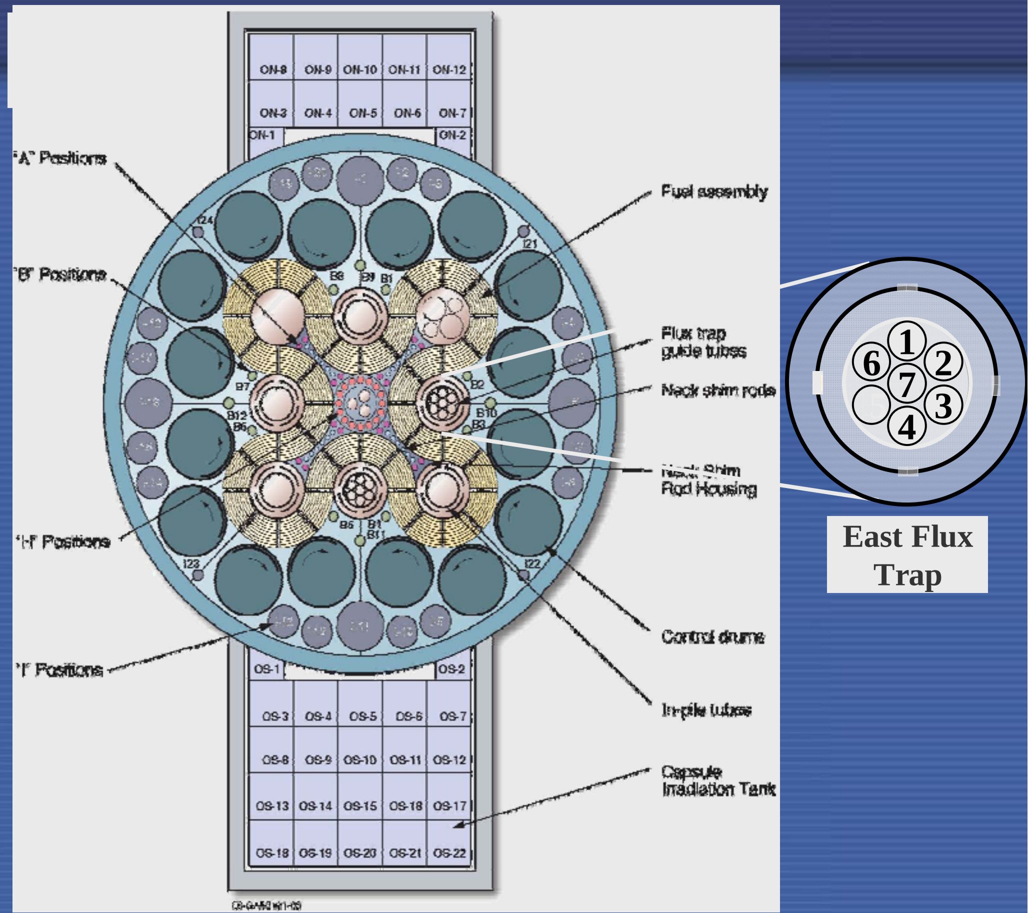
COST REDUCTION AND SAFETY R&D

- Nuclear Data
  - Advanced SFR requires new data and unprecedented precision to fully optimize the performance and economy of the system
  - Fission and capture measurements at LANSCE
- Materials
  - Alloy improvement beyond HT-9 and 316SS (NF616, TMT Modified NF616, HT-UPS, NF709)

COST REDUCTION AND SAFETY R&D, cont'd

❑Advanced Fuel (Recycle/Transmutation) Development: Modelling, Simulation, Fabrication, Characterization, Irradiation, PIE

- •Medium and high burnup irradiations in ATR
- •Two metallic and two nitride fuel rodlets fabricated and shipped to France for irradiation in Phénix (FUTURIX-FTA)



INL Advanced Test Reactor

Courtesy Ed Fujita, ANL

COST REDUCTION AND SAFETY R&D, cont'd

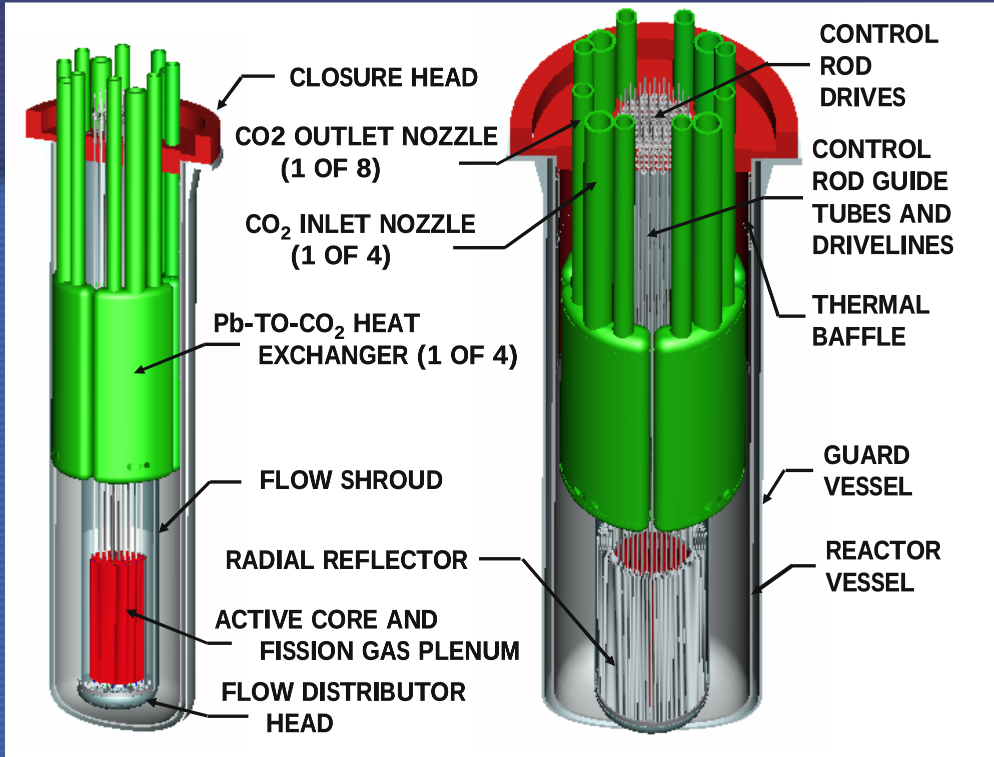
- Engineering
  - Sodium plugging tests
- Energy Conversion
  - Testing of super-critical Brayton CO_2 cycle using a small-scale (~1 MWth) facility with 50 kW centrifugal compressor

COST REDUCTION AND SAFETY R&D, cont'd

- Inherent Safety Evaluations
  - Review reactor-scale test data (EBR-II, FFTF, Monju, Phénix)
  - Review existing validation results and ongoing validation analyses using EBR-II, Monju, and Phénix data
  - Ongoing validation benchmarks using Phénix End-of-Life test and Monju start-up tests (IAEA Coordinated Research Projects)
- Severe Accident Approach
  - Design measures for prevention, e.g. self-actuated shutdown systems, gas expansion modules, core restraint systems for negative power coefficient, control rod driveline thermal expansion
  - Design measures for mitigation, e.g. filtered vented containment, reactor containment dome, JAEA FAIDUS fuel assembly, in-vessel core catcher
  - Review of the JAEA) and CEA severe accident safety approach
  - Performance evaluation and assessment of fast recycle reactor design features

LFR ACTIVITIES

- Small Secure Transportable Autonomous Reactor (SSTAR)
- Euratom Projects (ELSY and ETPP)



Courtesy Dave Wade, ANL

Small Secure Transportable Autonomous Reactor (SSTAR): 20 MWe, natural circulation in primary lead loop, secondary super-critical CO_2 loop, direct Brayton power conversion cycle

STATUS OF GIF FAST REACTOR SYSTEM AND PROJECT ARRANGEMENTS

- SFR
  - System Arrangement: China, Euratom, France, Japan, Rep. of Korea, USA
  - Advanced Fuel Project Arrangement: JRC (Euratom), CEA (France), JAEA (Japan), KAERI (Rep. of Korea), DOE (USA)
  - Global Actinide Cycle International Demonstration (GACID) Project Arrangement: CEA (France), JAEA (Japan), DOE (USA)
  - Component Design (CD) and Balance of Plant (BOP) Project Arrangement: CEA (France), JAEA (Japan), KAERI (Rep. of Korea), DOE (USA)
  - Safety and Operation Project Arrangement: CEA (France), JAEA (Japan), KAERI (Rep. of Korea), DOE (USA)
  - System Integration and Assessment Project Arrangement: on-going discussions between EU, France, Japan, Rep. of Korea, and USA

STATUS OF GIF SYSTEM AND PROJECT ARRANGEMENTS (FAST REACTORS)

- GFR
  - System Arrangement: Euratom, France, Japan, Switzerland
  - Fuel and Core Materials Project Arrangement: Project discussion being finalized by JRC(EU), CEA (France), JAEA (Japan), and PSI (Switzerland)
  - Conceptual Design and Safety Project Arrangement: Signatures requested from JRC(EU), CEA (France), and PSI (Switzerland)

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Summary: Fast Reactor Development

- France:
  - ✓ Conducting tests of transmutation of long lived waste and use of Pu fuels at Phénix; end-of-life experiments in Phénix (end of 2009) shut-down
  - ✓ Designing 300-600 MWe Advanced Prototype ASTRID, commissioning 2020
- Japan:
  - ✓ MONJU restart planned for 2010
  - ✓ Planned Operation of JOYO experimental LMR
  - ✓ Fast Reactor Cycle Technology Development Project (FaCT): 2025 Prototype, 2050 Commercial FR
- India
  - ✓ Operating FBTR
  - ✓ Constructing 500 MWe Prototype Fast Breeder Reactor (commissioning 2011)
  - ✓ Construction Projects (6 PFBR by 2020)
- Russia:
  - ✓ Operating BN-600
  - ✓ Constructing BN-800
  - ✓ Developing other Na, Pb, and Pb-Bi cooled systems
- China:
  - ✓ Constructing 25 MWe CEFR – criticality planned in 2010
- Rep. of Korea:
  - ✓ Conceptual design of 600 MWe Kalimer is complete
  - ✓ DEMO GEN-IV FR by 2030
- United States
  - ✓ SFR Preferred Option
  - ✓ Objective: Develop and Demonstrate Advanced Fast Recycle Reactor With Closed Fuel Cycle

Fast Reactors Looking Ahead …

- ***Renewed interest in nuclear energy***
- ***Sustainability ⇒ spent fuel utilization and breeding returning to centre stage ⇒ fast reactor necessary linchpin***
- ***Fast reactor deployment likely to be accelerated***
  - **Restart in 2010 of the industrial prototype Monju (Japan)**
  - **Commissioning, at the time horizon 2011 – 2023, of power fast reactors in India (PFBR series)**
  - **Planned construction by 2020 of the prototype fast reactor ASTRID (France)**
  - **Construction projects in India, Russia, Japan and the Republic of Korea**

Fast Reactors Looking Ahead …

- ***Necessary condition for successful deployment ⇒ understanding and assessment of technological and design options (based on past knowledge and experience, as well as on renewed research and technology development efforts)***
- ***Since 1967, IAEA's TWG-FR is an established collaboration framework assisting Member State fast reactor development and deployment activities by providing an umbrella for knowledge preservation, information exchange, and collaborative R&D***

For more information, please visit

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

Thank You !

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