GEN IV International Forum[SM]

SODIUM COOLED FAST REACTORS (SFR)

Dr Robert Hill

Argonne National Laboratory, USA

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NEA Nuclear Energy Agency

cea

Canadian Nuclear Laboratories

Laboratoires Nucléaires Canadiens

NATIONAL NUCLEAR LABORATORY

Argonne NATIONAL LABORATORY

INL Idaho National Laboratory

Berkeley UNIVERSITY OF CALIFORNIA

NPS PRAESTANTIA PER SCIENTIAM

Meet the presenter

Robert Hill is a senior Nuclear Engineering at Argonne National Laboratory, where he has worked for the last 29 years with research focused on reactor physics, fast reactor design, and fuel cycle applications.

Dr. Hill completed his Ph.D. in Nuclear Engineering at Purdue University in 1987. His current position at Argonne is Technical Director for Advanced Nuclear Energy R&D. He has previously led Nuclear Engineering Division research groups working on reactor physics analysis, advanced modeling and simulation, fuel cycle and system dynamic modeling, criticality safety, and nuclear data.

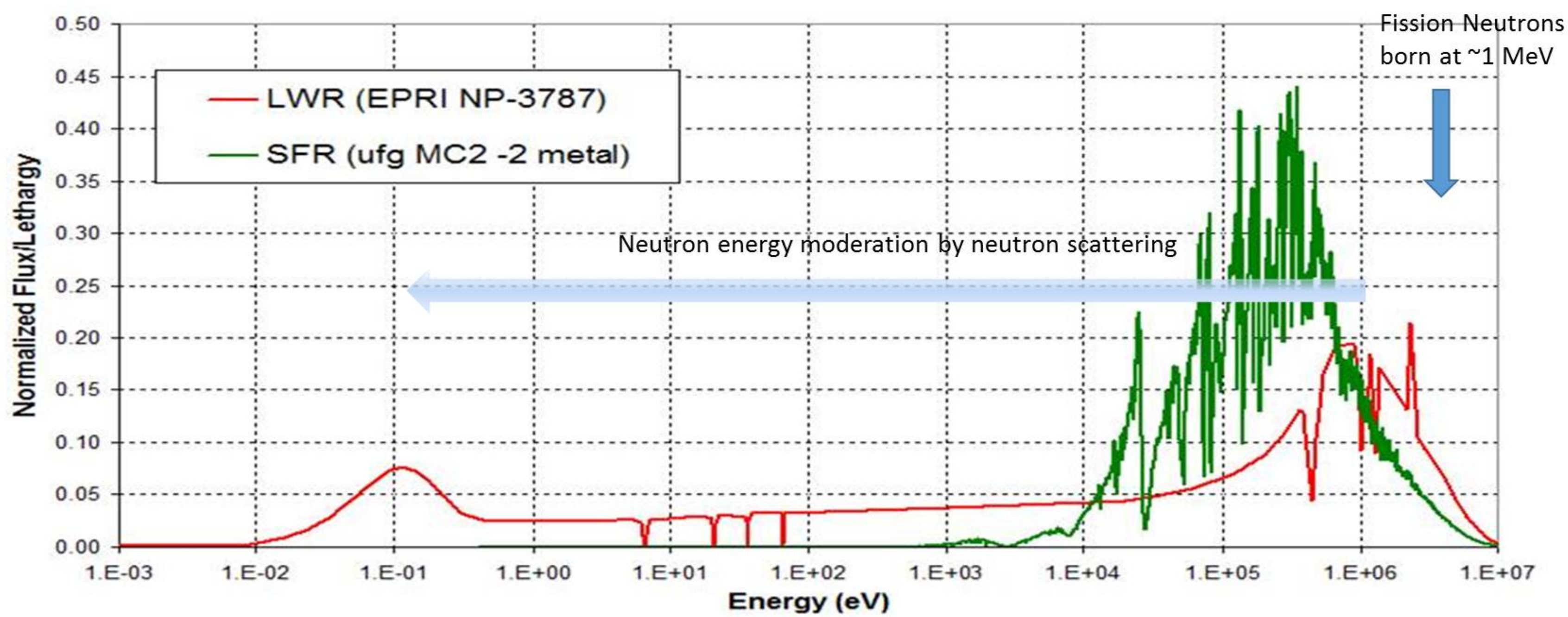
Robert Hill is co-National Technical Director for multi-Laboratory R&D activities in the DOE Advanced Reactor Technologies Program; this work includes small modular reactors, advanced structural materials, energy conversion technology, methods validation, non-LWR licensing, and system integration. He also serves as U.S. representative for the Generation-IV Sodium Cooled Fast Reactor collaboration.

Email: bobhill@anl.gov

Outline

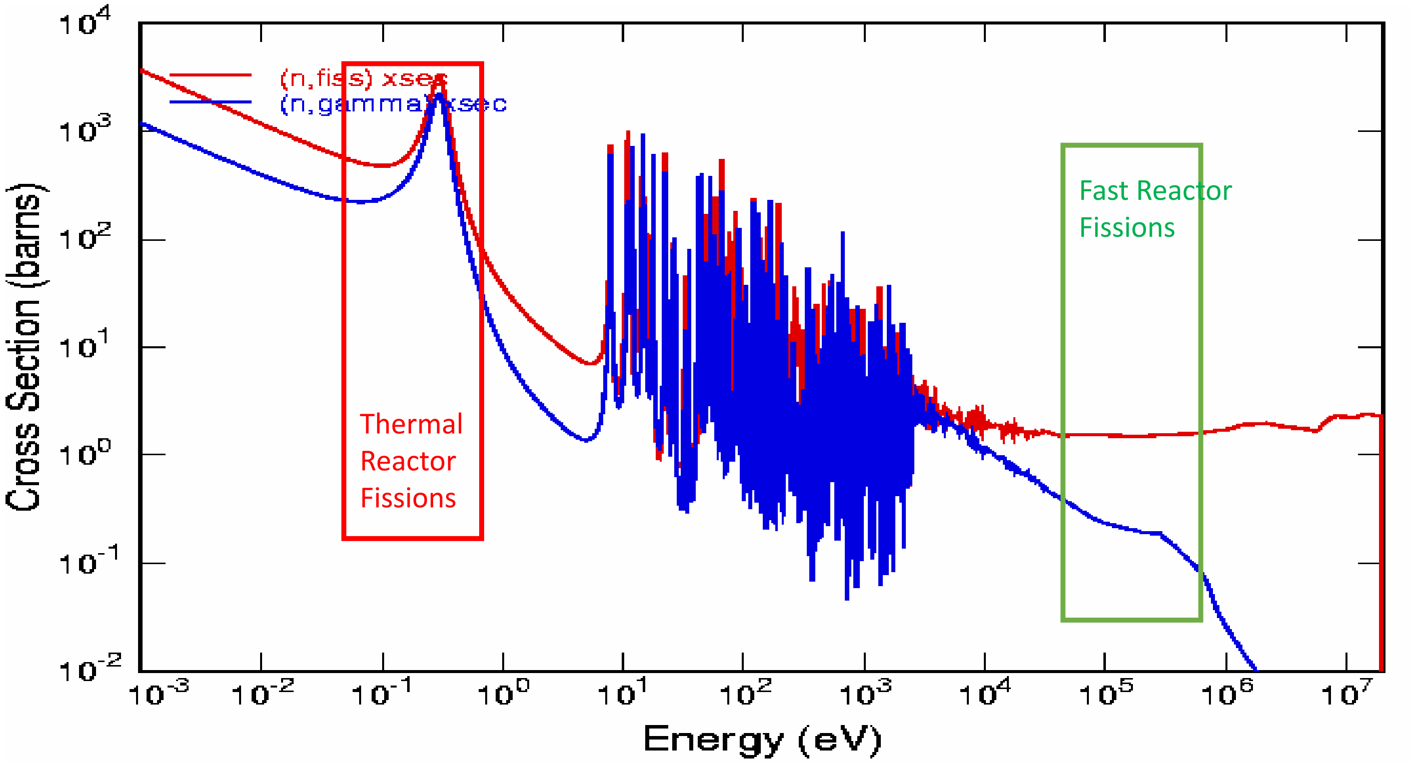
- **Fast Reactor Characteristics –** ***Role in Advanced Fuel Cycles***
- **Sodium Fast Reactor (SFR) Overview –** ***International Experience Base***
- **Generation-IV R&D Collaboration on SFR**

Fast Neutrons



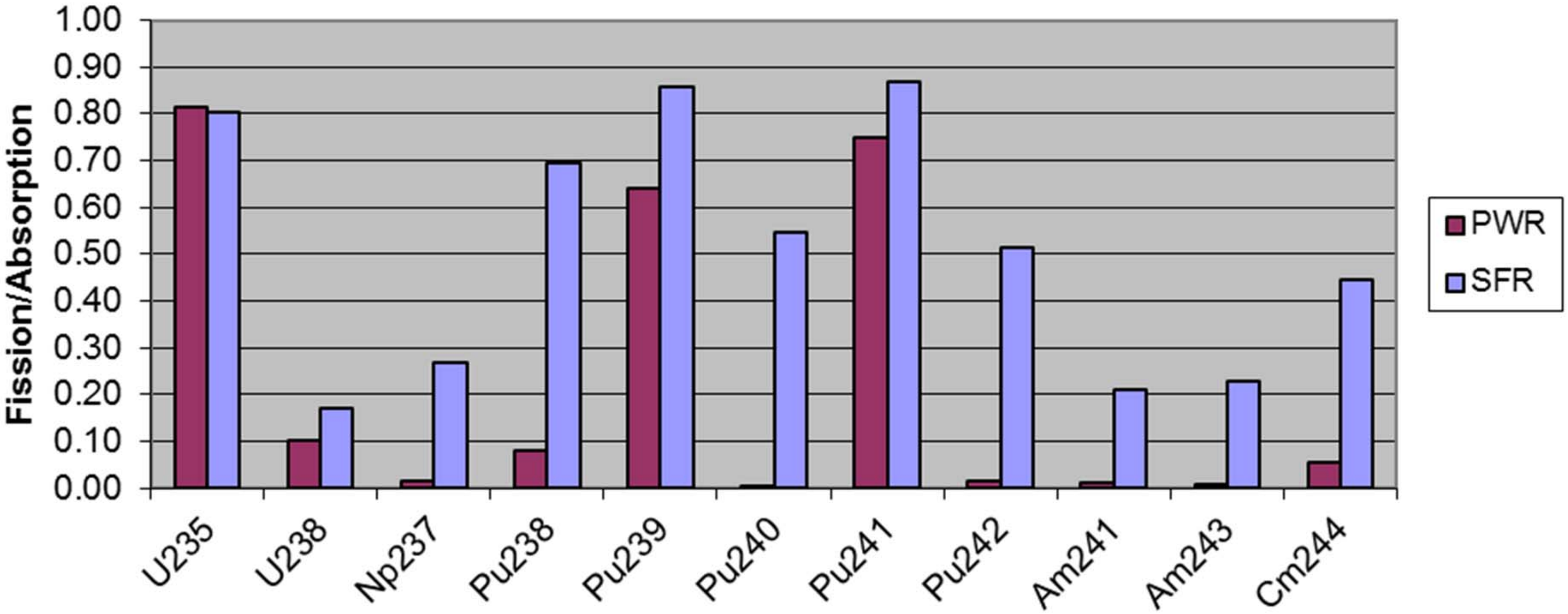
- **In LWRs, most fissions occur in the ~0.1 eV "thermal" peak**
- **In SFRs, neutron energy moderation is avoided – fission in "fast" energy range**

Pu-239 Fission and Capture



- Fission and capture cross section >100X higher in thermal range
- Sharp decrease in capture cross section at high energy

Impact of Energy Spectrum



- Fissile isotopes are likely to fission in both thermal/fast spectrum
  - Fission fraction is higher in fast spectrum
- Significant (up to 50%) fission of fertile isotopes in fast spectrum

Net result is more excess neutrons and less higher actinide generation in Fast Reactor

Fuel Cycle Implications

The reactor spectral behavior leads to different fuel cycle strategies:

- **Thermal reactors** typically configured for once-through (open) fuel cycle
  - They can operate on low enriched uranium (LEU)
  - They require an external fissile feed (neutron balance)
  - Higher actinides must be managed to allow recycle
    - Separation of higher elements – still a disposal issue
    - Extended cooling time for curium decay
- **Fast reactors** are typically intended for closed (recycle) fuel cycle with uranium conversion and resource extension
  - Higher actinide generation is suppressed
  - Neutron balance is favorable for recycled transuranics (Pu, Np, and Am)
    - No external fissile material is required
    - Can enhance U-238 conversion for traditional breeding
    - Can limit U-238 conversion for burning

Vision for Fast Reactors

From the initial conception of nuclear energy, it was recognized that full realization of uranium energy content would require fast reactors

Fermi: The vision to close the fuel cycle

↓

50's: First electricity generating reactor: EBR-I with a vision to close the fuel cycle for <u>resource extension</u>

↓

60-70's: Expected Uranium scarcity –significant Fast Reactor programs

↓

80's: Decline of nuclear – Uranium plentiful

2 paths

→ USA (& others): once through cycle & repository

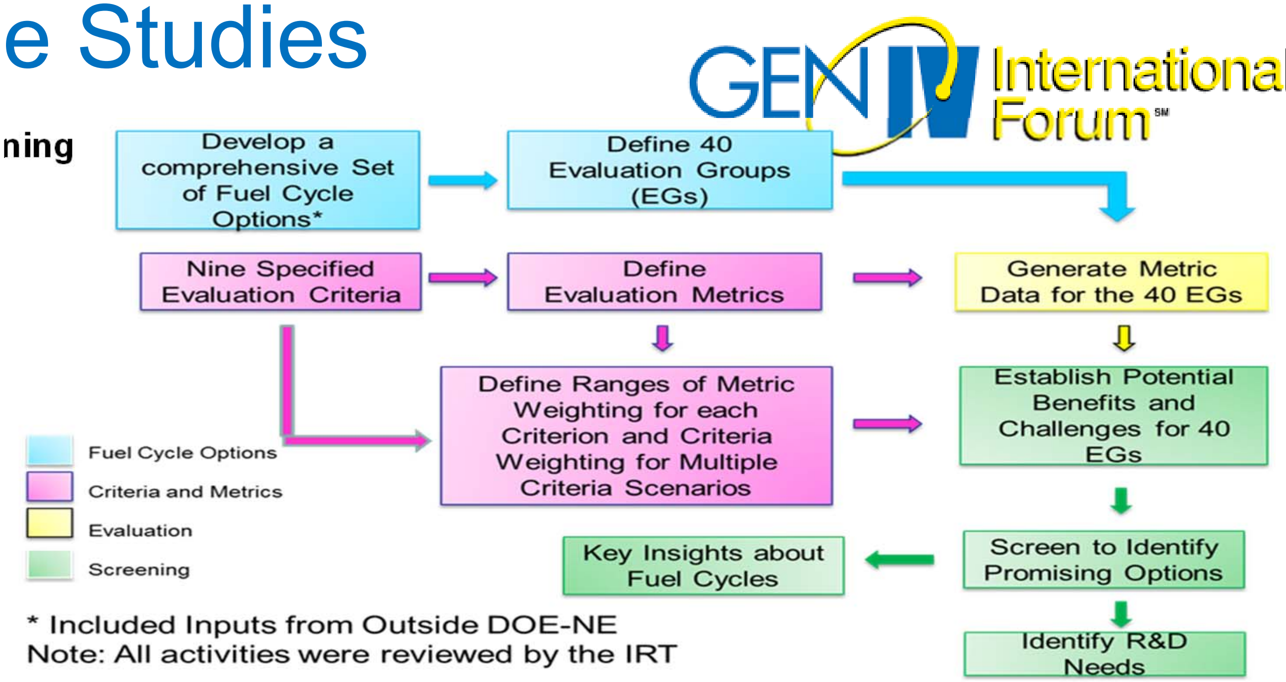
→ France, Japan (& others): closed cycles to mitigate and delay waste disposal

Late 90's in the U.S.: Rebirth of closed cycle research and development for improved <u>waste management</u>

Now: flexible actinide management for fuel cycle missions

Recent Fuel Cycle Studies

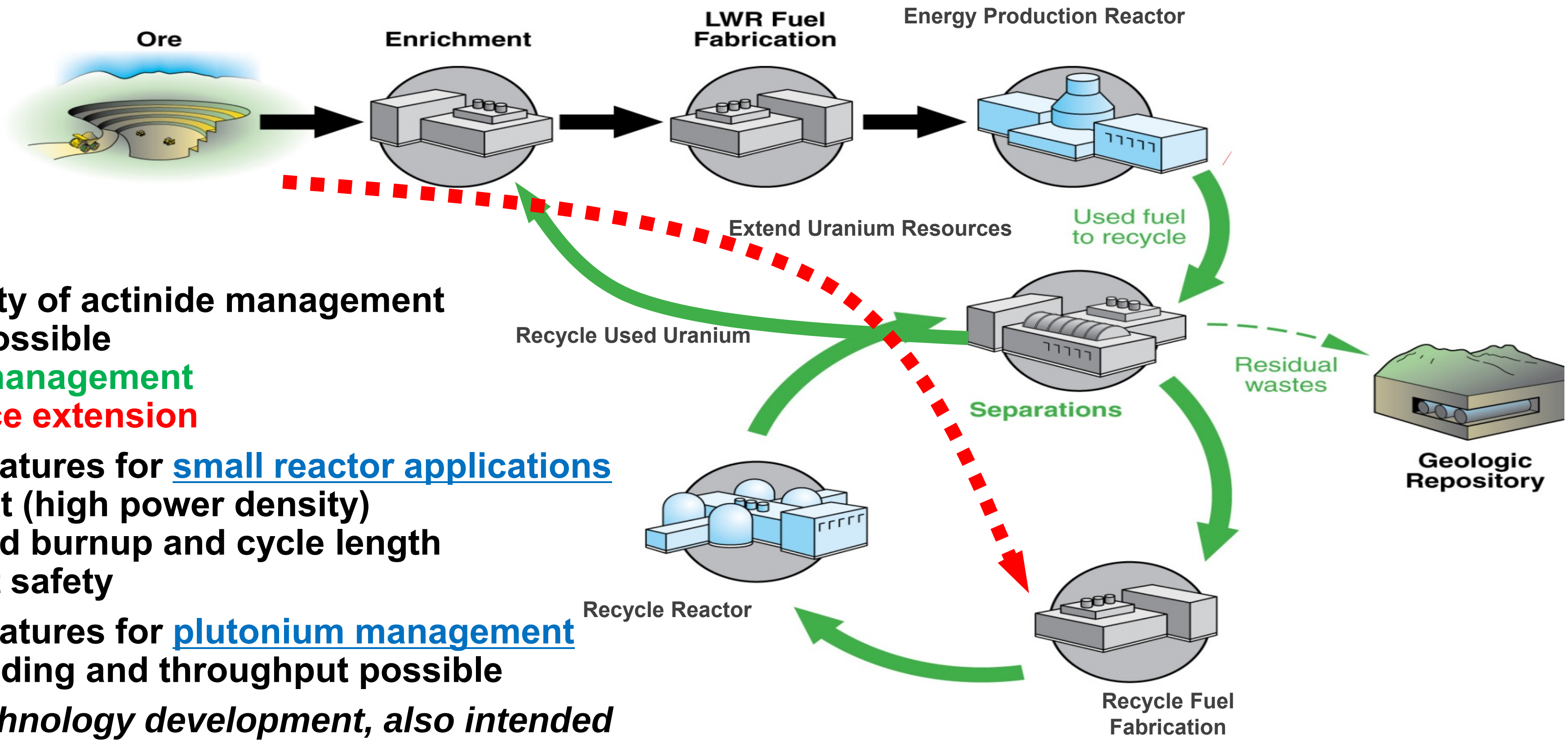
- **Consider complete fuel cycle from mining to disposal, in deployed steady-state**
- **Develop a <u>comprehensive</u> set of fuel cycles with respect to performance**
  - *Technology-neutral*
  - *~4400 groups into 40 evaluation groups*
- **Develop evaluation criteria/metrics**
- **Explore impacts of different criteria weighting factors that reflect policy**



- For the Criteria and Metrics used in the Evaluation and Study, the best performing fuel cycles have one or more of the following characteristics
  - Continuous recycle of actinides (U/Pu or U/TRU)
  - Fast neutron-spectrum critical reactors
  - High internal conversion (of fertile to fissile)
  - No uranium enrichment is required once steady-state conditions are established

Results confirm that fast spectrum systems have a vital role for improved performance of advanced fuel cycles

Actinide Management in Fast Reactors



- **A wide variety of actinide management strategies possible**
  - **Waste management**
  - **Resource extension**
- **Favorable features for small reactor applications**
  - **Compact (high power density)**
  - **Extended burnup and cycle length**
  - **Inherent safety**
- **Favorable features for plutonium management**
  - **High loading and throughput possible**
- ***With key technology development, also intended for electricity, heat production, or other energy product missions***

Uranium Utilization

Once-through systems

	PWR-50GWd/t	PWR-100GWd/t	VHTR	Fast Burner
Burnup, %	5	10	10.5	22.3
Enrichment, %	4.2	8.5	14.0	12.5
Utilization, %	0.6	0.6	0.4	0.8

Recycling Systems

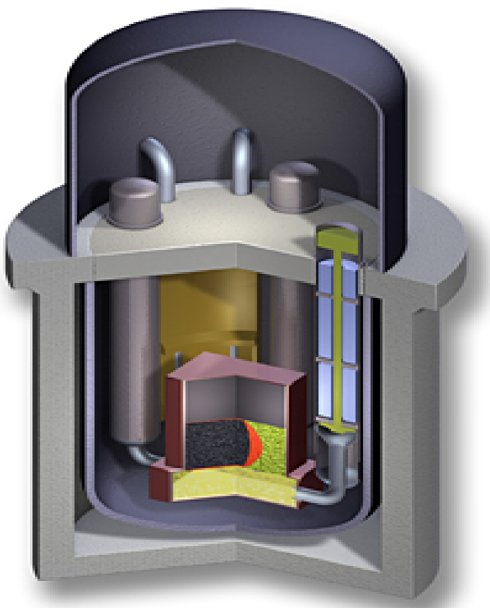
	LWR		LWR-Fast Burner		Fast
	UOX	MOX	LWR-UOX	Fast Burner	Converter
Power sharing, %	90	10	57	43	100
Burnup, %	5	10	5	9	-
Enrichment, %	4.2	-	4.2	12.5	-
Utilization, %	0.7		1.4		~99

Is it possible to improve U utilization significantly …

- **without recycle?**
- **with limited recycle?**

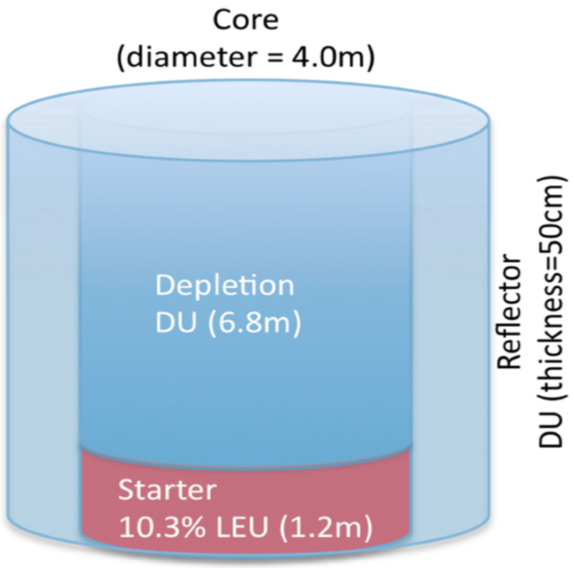
Breed and Burn (BB) Principles

- **Enriched U-235 (or Pu-239) starter core would be surrounded by a blanket of fertile fuel**
- **Enriched fuel would produce neutrons that generate power and convert fertile fuel to fissionable fuel**
- **Irradiated fertile fuel would replace enriched fuel after original U-235 (or Pu-239) is burned and new Pu-239 is formed**
- **Use of "Standard Breeders" exploit this physics in conjunction with reprocessing**
  - Complete U-238 conversion and fission, with the uranium utilization limited only by losses
- **Breed and Burn concepts promote conversion, but minimize reprocessing**
  - Once fertile zone dominates, once-through uranium utilization at the fuel burnup limit

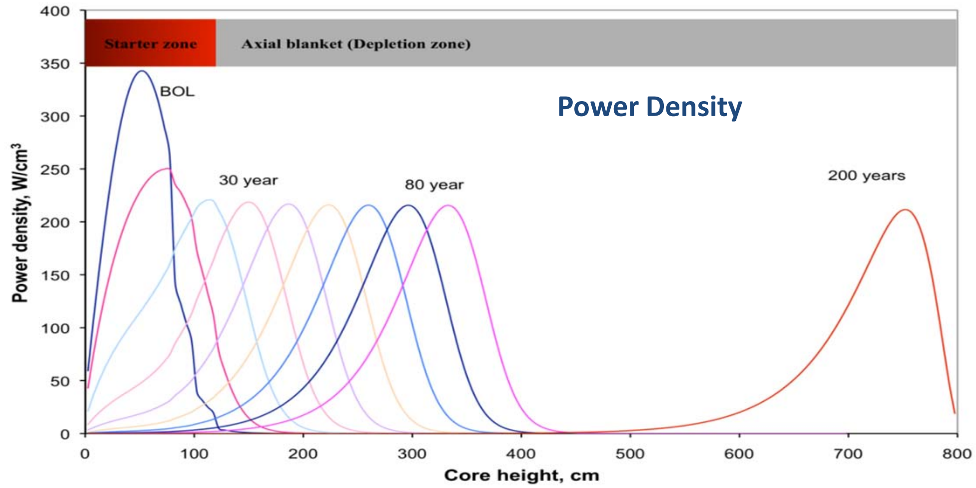


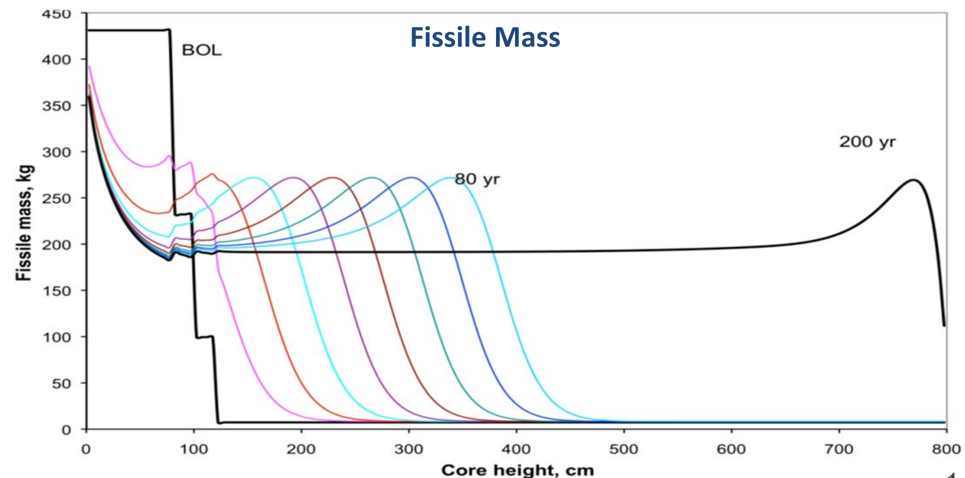
Travelling Wave Concept

BB Example - CANDLE



- Fissioning zone propagates from starter into depleted uranium (DU) region
- In principle, reactor operation can be extended in proportion to height of the DU region





BB Fuel Cycle Performance

Parameter	PWR-50/100	CANDLE	Conventional SFR
Reactor power, MWt	3000	3000	840
Thermal efficiency, %	33.3	40.0	40.0
Fuel Form	UO_2	U-Zr	U-TRU-Zr
Fissile enrichment, %	4.2 / 8.5	1.2	20
Number of batches	3	1	3
Burnup, GWd/t	50 / 100	**258**	100
Specific power density, MW/t	33.7	3.7	37
Cycle length per batch, yr	1.5 / 3.0	[a] 200	2
HM inventory, t	89	**824**	23
HM fission, t/yr	1.03	1.03	1.03

a) **Reactor operation time with 8 m active core height**

- Compared to conventional systems, once-through fast reactor systems have high initial heavy metal (HM) loading (significantly lower specific power density)
- Due to long fuel residence time, average burnups higher than that of conventional PWRs or SFRs; *however, much higher neutron damage must be tolerated*

Historical Perspective on Reactor Coolants

- **In the 1950s and 1960s, scientists and engineers considered (and in many cases built) nearly everything imaginable at the time:**
  - Water (light, heavy)
  - Liquid-metal (NaK, sodium, lithium, mercury, rubidium, lead, bismuth, lead-bismuth, gallium, tin, etc. and numerous other alloys)
  - Gas (air, argon, carbon dioxide, helium, hydrogen, nitrogen)
  - Fluid Fuel (aqueous: UO_2/phosphoric acid, $U(SO_4)_2$, UO_2SO_4/ThO_2; molten salt: $NaF\text{-}BeF_2\text{-}UF_4$, $LiF\text{-}BeF_2\text{-}ZrF_4\text{-}UF_4$/FLiBe; molten metal: U-Bi, Pu-10Fe)
  - Organic (polyphenyls/terphenyls, kerosene, Santowax)
- **Combinations of coolant and moderator were also studied:**
  - Sodium-cooled, graphite moderated (SRE, Hallam)
  - Organic-cooled, heavy water moderated (Whiteshell 1, ESSOR)

Sodium as a Fast Reactor Coolant

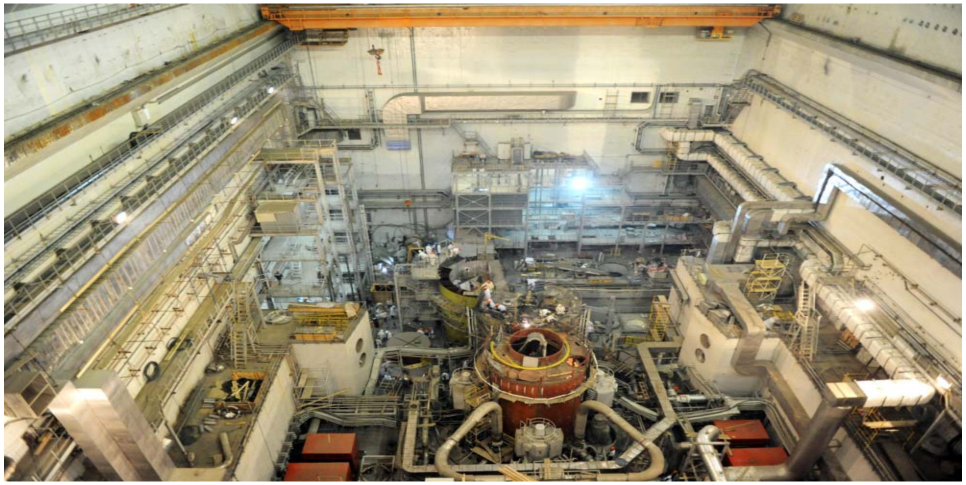
- Thermophysical and thermal-hydraulic properties of sodium are excellent and allow:
  - Use of conventional stainless steels
  - Smaller core with higher power density, lower enrichment, and lower heavy metal inventory
  - Demonstrated natural circulation and overall passive safety performance
  - Use of sodium codified in ASTM standards
- Extensive testing resulted in sodium as the primary coolant in nearly all (land-based) fast reactors constructed during the last 50 years.
  - Current fast reactor construction projects use sodium as the primary coolant
  - LBE-cooled reactors limited to Russian Alfa-class submarine experience

Thermophysical Properties:		
	Excellent Heat Transfer	✓+
	Low Vapor Pressure	✓+
	High Boiling Point	✓+
	Low Melting Point	✓
Material Properties:		
	Thermal Stability	✓+
	Radiation Stability	✓+
	Material Compatibility	✓+
Neutronic Properties:		
	Low Neutron Absorption	✓+
	Minimal Activation	✓
	Negligible Moderation	✓+
Supports Passive Safety		✓+
Cost:		
	Initial Inventory	✓+
	Make-Up Inventory	✓+
	Low Pumping Power	✓+
Hazards:		
	Sodium reacts with air and water	

International Fast Reactors

Reactor	Country	MWth	Operation	Coolant
EBR 1	**USA**	**1.4**	**1951-63**	**NaK**
BR-2	**Russia**	**2**	**1956-1957**	**Mercury**
BR-10	**Russia**	**8**	**1959-71, 1973-2002**	**Sodium**
DFR	**UK**	**60**	**1959-77**	**NaK**
EBR II	**USA**	**62.5**	**1963-94**	**Sodium**
Fermi 1	**USA**	**200**	**1963-72**	**Sodium**
Rapsodie	**France**	**40**	**1966-82**	**Sodium**
BOR-60	**Russia**	**50**	**1968-**	**Sodium**
SEFOR	**USA**	**20**	**1969-1972**	**Sodium**
OK-550/BM-40A	**Russia**	**155 (7 subs)**	**1969-**	**Lead Bismuth**
BN 350*	**Kazakhstan**	**750**	**1972-99**	**Sodium**
Phenix	**France**	**563**	**1973-2009**	**Sodium**
PFR	**UK**	**650**	**1974-94**	**Sodium**
KNK 2	**Germany**	**58**	**1977-91**	**Sodium**
Joyo	**Japan**	**140**	**1978-**	**Sodium**
FFTF	**USA**	**400**	**1980-93**	**Sodium**
BN 600	**Russia'**	**1470**	**1980-**	**Sodium**
Superphenix	**France**	**3000**	**1985-98**	**Sodium**
FBTR	**India**	**40**	**1985-**	**Sodium**
Monju	**Japan**	**714**	**1994-96, 2010-**	**Sodium**
CEFR	**China**	**65**	**2010-**	**Sodium**
PFBR	**India**	**1250**	**2016?**	**Sodium**
BN-800	**Russia**	**2000**	**2014-**	**Sodium**
ASTRID	**France**	**1500**	**2025?**	**Sodium**
PGSFR	**Korea**	**400**	**2028**	**Sodium**

Fast Reactor Experience



Early experience

- First usable nuclear electricity was generated by a fast reactor – the EBR-I in 1951
- EBR-II (20 MWe) was operated at Idaho site from 1963 to 1994
  - Metal and oxide fuels demonstrated
  - Inherent safety testing in 1986
- FERMI-1 fast commercial power reactor (61 MWe) in 1965

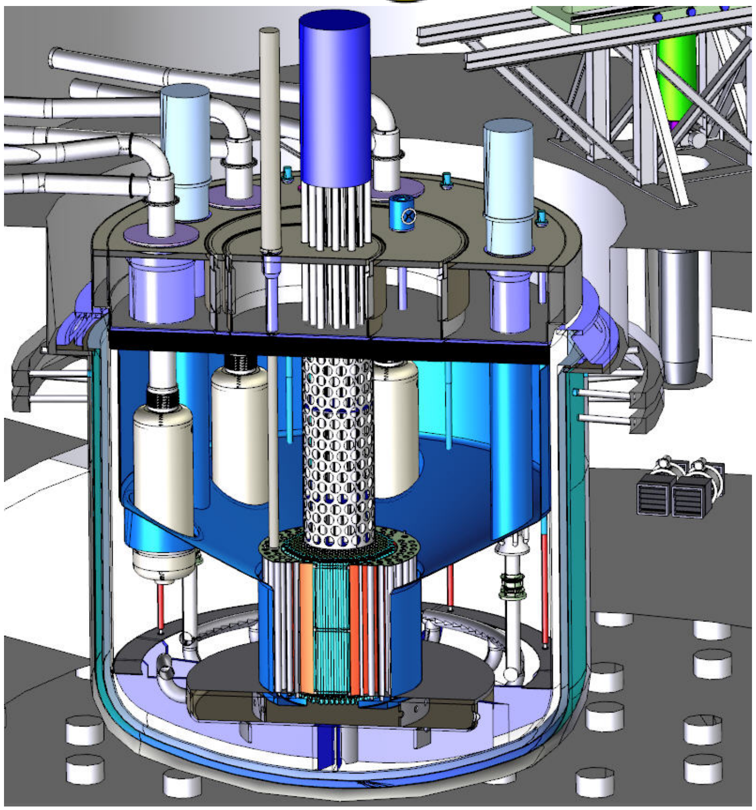
Worldwide Experience

- About 20 fast reactors with 400 operating-years
- Test and/or demonstration reactors built and operated in US, France, UK, Russia, Japan, India, Germany, and China
- Recent startup of CEFR and restart of MONJU
- New power reactors: PFBR (500 MWe) and BN-800 (880 MWe)

Basic viability of sodium-cooled fast reactor technology is demonstrated

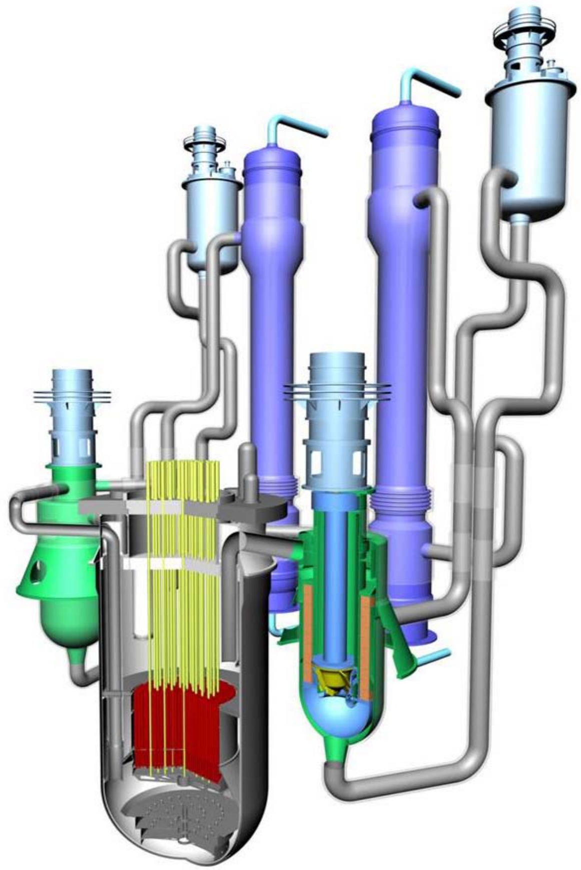
Pool Configuration

- All components of the primary system are immersed in sodium coolant contained within one main tank
- Hot-pool – PFR, Phenix/SuperPhenix, BN-600/800
  - Much of pool top surface and significant part of the main-tank contents are at the reactor coolant outlet temperature
  - Early designs (EBR-II) used cold pool
- Some advantages
  - Radioactive sodium is located in one tank
  - Primary sodium boundary is simple geometry
  - Thermal inertia and circuit arrangements make the primary circuit more tolerant of transients and component failure.
  - Smaller containment building (in general)
  - Can better tolerate minor leakage within the primary tank system



Loop Configuration

- The individual components of the primary heat transport system such as the reactor vessel, IHX, primary pump and check valves connected to each other by a piping system
- The piping and components are arranged to form a continuous loop to the IHX where heat is transferred to the secondary coolant
- Utilized in FFTF, Monju (CRBR and JSFR designs)
- Some advantages
  - More freedom of choice for optimization and location of components
  - Short span reactor vessel cover with only a few penetrations in a blanket gas space that is isothermal
  - Simple boundary between hot and cold parts of the circuit
  - Support of the core requires a smaller structure because of the short span
  - IHX's can have a higher pressure drop which eases demonstration of satisfactory flow distribution



Intermediate Loop

All existing SFR reactors, and most modern designs utilize an intermediate sodium loop, which allows:

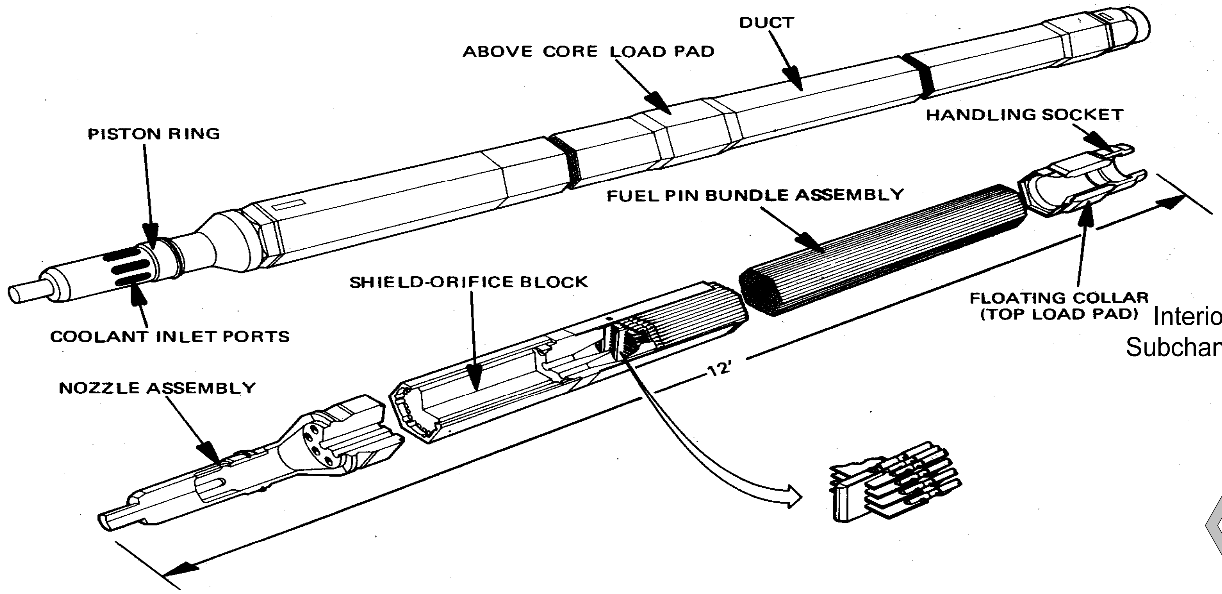
- Isolation of the primary loop from the steam generators
  - No impact on core for failures in steam generator
  - Historical concern for sodium-water SG reliability
- Keep reactor vessel components at low pressure
- Steam generators reside outside the containment

Elimination of the intermediate loop has been identified as potential for cost reduction, but raises the noted design issues

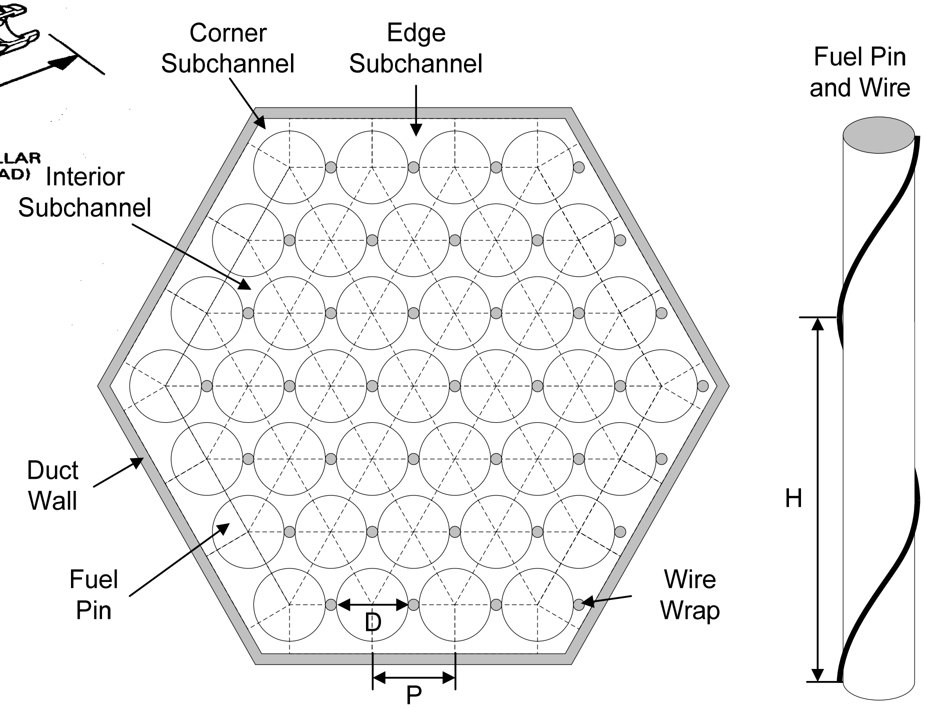
Typical Operating Conditions

- Sodium cooled fast reactors typically operate at near atmospheric pressure; peak pressures are set by core pressure drop and gravity head characteristics (up to about 1.0 MPa max at reactor inlet)
- Reactor coolant outlet temperatures are typically around 510°C to 550°C, depending on cladding material (margin to boiling 330°C to 370°C)
- Average power densities in the reactor core are typically 300 to 500 kW/liter, or about 1100 to 1500 kW/liter in the fuel
- Average fuel pin linear power ratings are typically 23 to 28 kW/m for pins with cladding diameters of 6 to 8 mm
- *Fuel pins are typically arranged on a triangular pitch, and positioned by a spiral wire spacer within a hexagonal assembly duct*
- *Typical coolant velocities in the fuel pin bundle are 5 to 7 m/s*
- *At reactor temperatures, sodium wets stainless steel, which is typically used as the cladding and structural material*
- *Each reactor fuel assembly typically produces about 5 MW of power*

Typical SFR Assembly Design

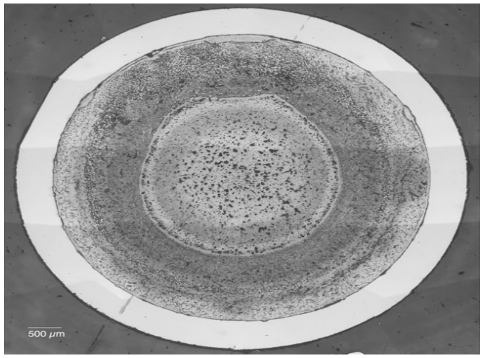


Fuel Assembly (FFTF)

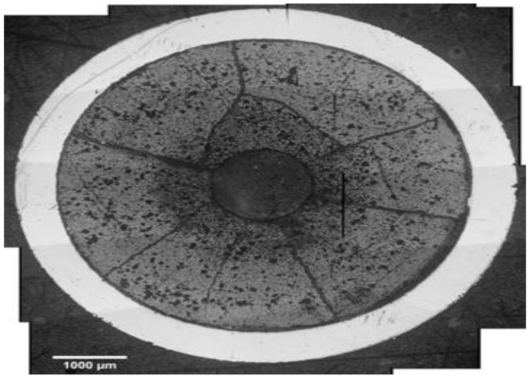


Metal Alloy and Oxide Fuel Properties

	Metal (U-20Pu-10Zr)	Oxide (UO_2-20PuO_2)
Heavy Metal Density, g/cm^3	**14.1**	**9.3**
Melting Temperature, °K	**1375**	**3000**
Thermal Conductivity, W/cm-°K	**0.2-0.3**	**0.02-0.04**
Operating Centerline Temp. at 40 kW/m, °K, and (T/T_{melt})	**1060 (0.77)**	**2360 (0.79)**
Fuel-Cladding Solidus, °K	**1000**	**1675**
Thermal Expansion, 1/°K	**17×10^{-6}**	**12×10^{-6}**
Heat Capacity, J/g°K	**0.2**	**0.3-0.5**



Metal Fuel with HT9 Clad



High Burnup MOX Fuel

Design Issues for Fuel Options

- Fuel Swelling
  - Fission product retention in carbide and nitride fuels can lead to greater swelling than observed for oxide fuels and exacerbate FCMI
  - *Current metal and oxide fuel pin designs accommodate fuel swelling*
- Fuel / Cladding Chemical Interaction
  - Metal uranium and plutonium forms low-melting point eutectic with iron
  - May limit coolant outlet temperature of metal fuel core, e.g., 510°C for metal as compared to ~550°C for oxide (structural materials limiting)
- Fuel / Cladding Mechanical Interaction (FCMI)
  - Hard, strong fuel forms push on cladding, particularly at high burnup
  - Worst for nitride and carbide, limits maximum burnup for ceramic fuels
- Fuel / Coolant Compatibility
  - Oxide fuel chemically reacts with the sodium coolant
    - Stricter limits on fuel pin failures to prevent potential flow blockages
  - Lack of sodium bond inside the fuel pin results in low thermal conductivity between fuel and cladding and higher fuel temperatures

LWR and SFR Performance Specifications

			PWR	SFR
General	Specific power (kWt/kgHM)		786 (U-235)	556 (Pu fissile)
	Power density (MWt/m³)		102	300
Fuel	Rod outer diameter (mm)		9.5	7.9
	Clad thickness (mm)		0.57	0.36
	Rod pitch-to-diameter ratio		1.33	1.15
	Enrichment (%)		~4.0	~20 Pu/(Pu+U)
	Average burnup (MWd/kg)		40	100
Thermal Hydraulic	Coolant	pressure (MPa)	15.5	0.1
		inlet temp. (℃)	293	350
		outlet temp. (℃)	329	500
		reactor Δp (MPa)	0.345	0.827
	Rod surface heat flux	average (MW/m²)	0.584	1.1
		maximum MW/m²)	1.46	1.8
	Average linear heat rate (kW/m)		17.5	27.1
	Steam	pressure (MPa)	7.58	15.2
		temperature (℃)	296	455

Inherent Safety Approach

- Superior thermophysical properties of liquid metals allow:
  - Operation at high power density and high fuel volume fraction
  - Low pressure operation with significant margin to boiling
  - Passive decay heat removal
- The fast neutron spectrum leads to long neutron path lengths
  - Neutron leakage is enhanced (25% at moderate sizes)
  - Changes in power level effects impact the reactor as a whole, not locally
- High leakage fraction implies that the neutron balance is sensitive to minor geometric changes
  - As temperature increases and materials expand, a net negative reactivity feedback is inherently introduced
- Favorable inherent feedback in sodium-cooled fast reactors (SFR) have been demonstrated
  - EBR-II and FFTF tests for double fault accidents
  - Safety codes developed and validated to model the coupled physics, thermal, structural reactivity feedback effects

Generation-IV Goals

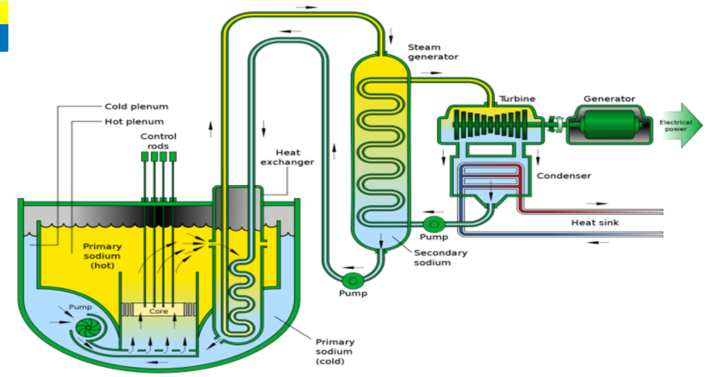
- Eight goals for the Generation IV nuclear energy systems have been defined in the four broad areas of sustainability, economics, safety and reliability, and proliferation resistance and physical protection

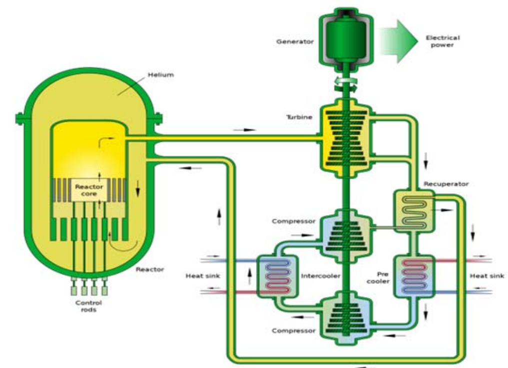
Criteria	Goal: Generation IV nuclear energy systems will….
Safety and Reliability-1	*excel in safety and reliability.*
Safety and Reliability-2	*have a very low likelihood and degree of reactor core damage.*
Safety and Reliability-3	*eliminate the need for offsite emergency response.*
Economics-1	*will have a clear life-cycle cost advantage over other energy sources.*
Economics-2	*will have a level of financial risk comparable to other energy projects.*
Sustainability-1	*will provide sustainable energy generation that meets clean air objectives and promotes long-term availability of systems and effective fuel utilization for worldwide energy production.*
Sustainability-2	*will minimize and manage their nuclear waste and notably reduce the long-term stewardship burden, thereby improving protection for the public health and the environment.*
Proliferation Resistance and Physical Protection-1	*increase the assurance that they are a very unattractive and the least desirable route for diversion or theft of weapons-usable materials, and provide increased physical protection against acts of terrorism.*

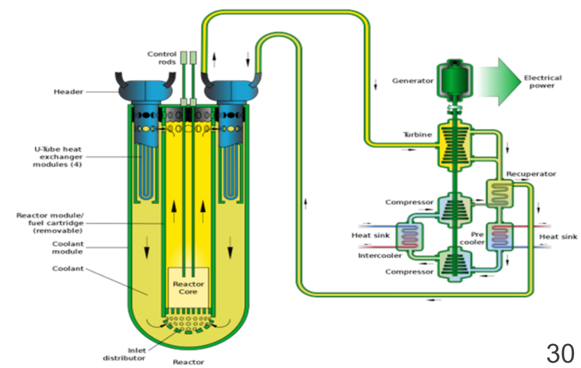
Generation IV Fast Reactors

- Generation IV International Forum was chartered in May 2001.
- “A Technology Roadmap for Generation IV Nuclear Energy Systems,” Generation IV International Forum, GIF-002-00, December 2002
- "Technology Roadmap Update for Generation IV Nuclear Energy Systems,” Generation IV International Forum, January 2014

- Roadmap identified <u>three fast reactor technologies </u>for consideration
  - Sodium-cooled fast reactor (SFR)
  - Lead (LBE) cooled fast reactor
  - Gas-cooled fast reactor (GFR)
  - *Also, fast version of SCWR and MSR options proposed*

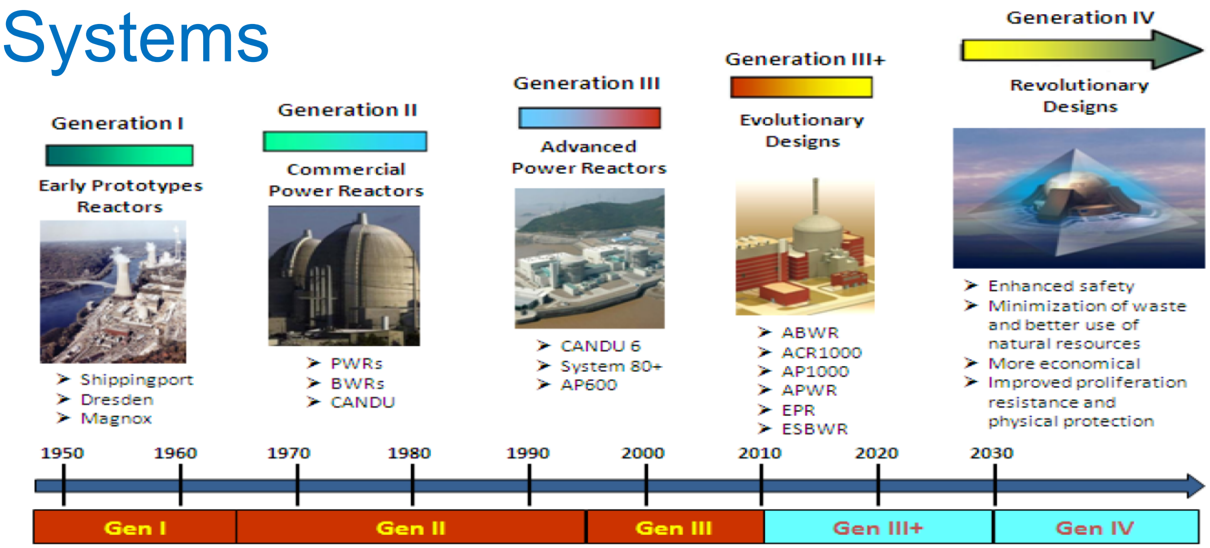






Generation IV Nuclear Systems

- Six Generation IV Systems considered internationally
- Often target missions beyond electricity
  - High temperature energy products
  - Fuel cycle benefits



System	Neutron spectrum	Coolant	Outlet coolant Temp. °C	Fuel cycle	Size (MWe)
VHTR (Very high temperature reactor)	thermal	helium	900-1 000	open	250-300
SFR (Sodium-cooled fast reactor)	fast	sodium	550	closed	30-150, 300-1 500, 1 000-2 000
SCWR (Supercritical water cooled reactor)	thermal/fast	water	510-625	open/closed	300-700 1 000-1 500
GFR (Gas-cooled fast reactor)	fast	helium	850	closed	1200
LFR (Lead-cooled fast reactor)	fast	lead	480-800	closed	20-180, 300-1 200, 600-1 000
MSR (Molten salt reactor)	Epithermal/fast	fluoride salts	700-800	closed	1 000

Basic Description of Gen IV SFR

- Mission
  - Primary mission for SFR is the improved utilization of uranium resources and effective management of high-level waste
  - With reduction of capital cost, SFR is an attractive option for electricity production

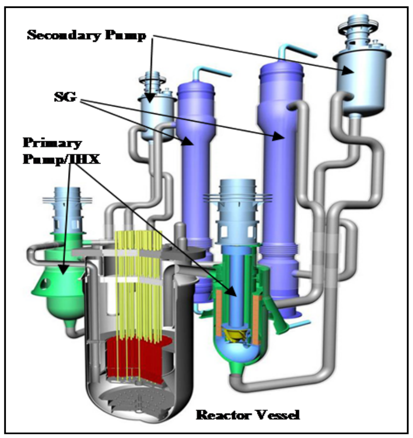
- Typical Design Parameters for Generation IV SFR

- Outlet Temperature	500-550 ^{0}C
- Power Rating	50-2,000 MWe
- Fuel	Oxide, Metal Alloy
- Breeding Ratio	0.5 – 1.3

Generation IV SFR Design Tracks

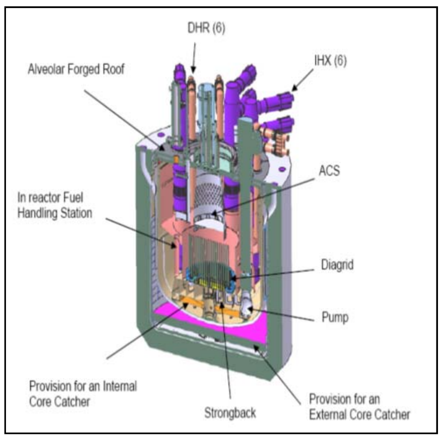
Loop

JSFR

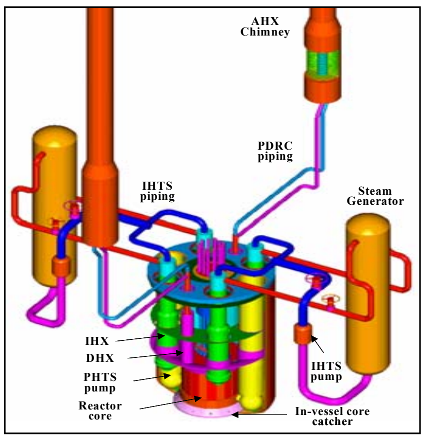


Pool

ESFR

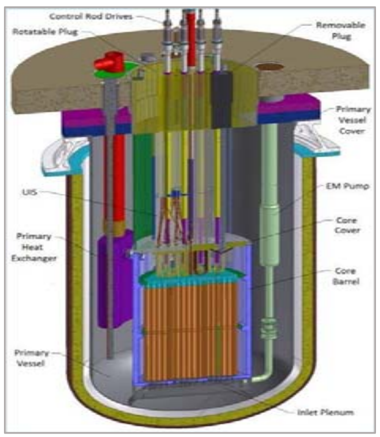


KALIMER



Small Modular

AFR-100



SFR System Research Plan

Development Targets and Design Requirements

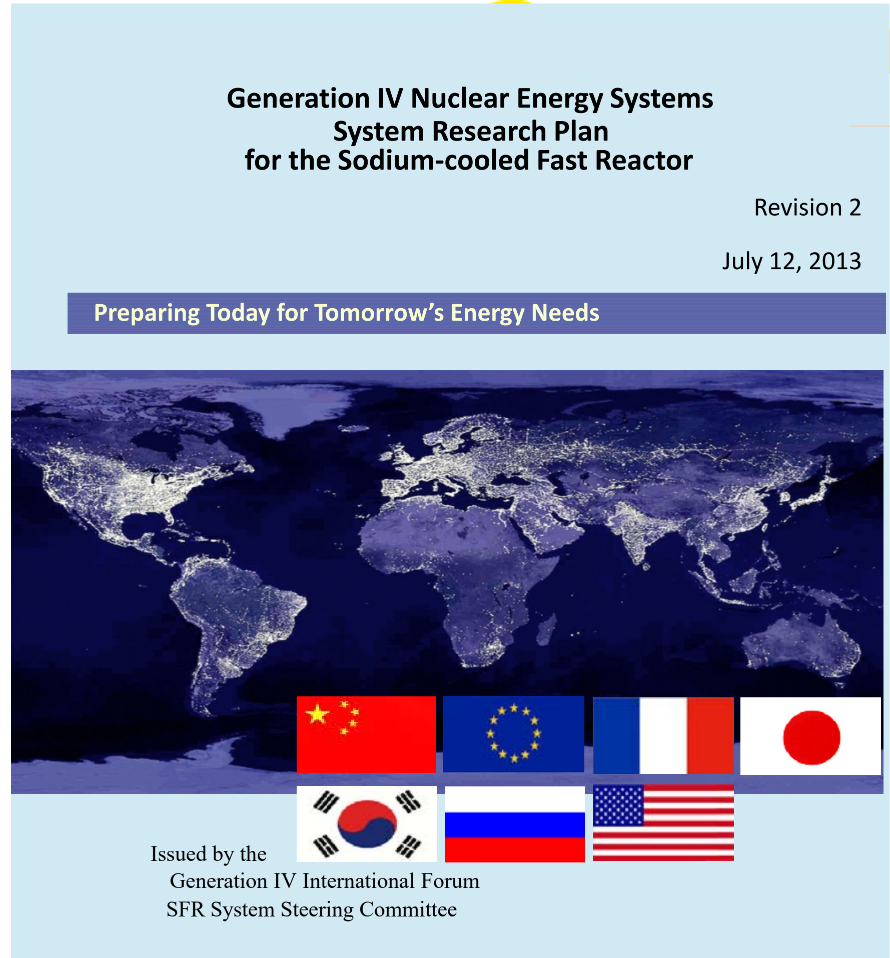
5 SFR R&D Projects

- System Integration and Assessment (SIA)
- Safety and Operations
- Advanced Fuel
- Component Design and Balance of Plant (CD & BOP)
- Global Actinide Cycle International Demonstration (GACID)

4 SFR Design Concepts

- Loop Option (JSFR Design Track)
- Pool Option (KALIMER-600 & ESFR Design Tracks)
- Small Modular Option (AFR-100 Design Track)

System Research Plan was updated and released in July 2013



Generation IV Nuclear Energy Systems
System Research Plan
for the Sodium-cooled Fast Reactor

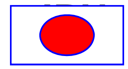
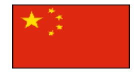
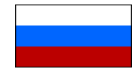
Revision 2

July 12, 2013

Preparing Today for Tomorrow's Energy Needs

Issued by the
Generation IV International Forum
SFR System Steering Committee

Status of SFR Arrangements

							
SFR System Arrangement **(15 Feb 2006, extend Feb 2016)**	X	X	X	X	X	X	X
SFR AF PA **(21 Mar 2007)**	X	X	X	X	X	X	X
SFR GACID PA **(Sept 2007)**		X	X				X
SFR CDBOP PA **(11 Oct 2007)**	D	X	X	D	X	D	X
SFR SO PA **(11 June 2009)**	X	X	X	X	X	X	X
SFR SIA PA **(22 Oct 2014)**	X	X	X	X	X	X	X

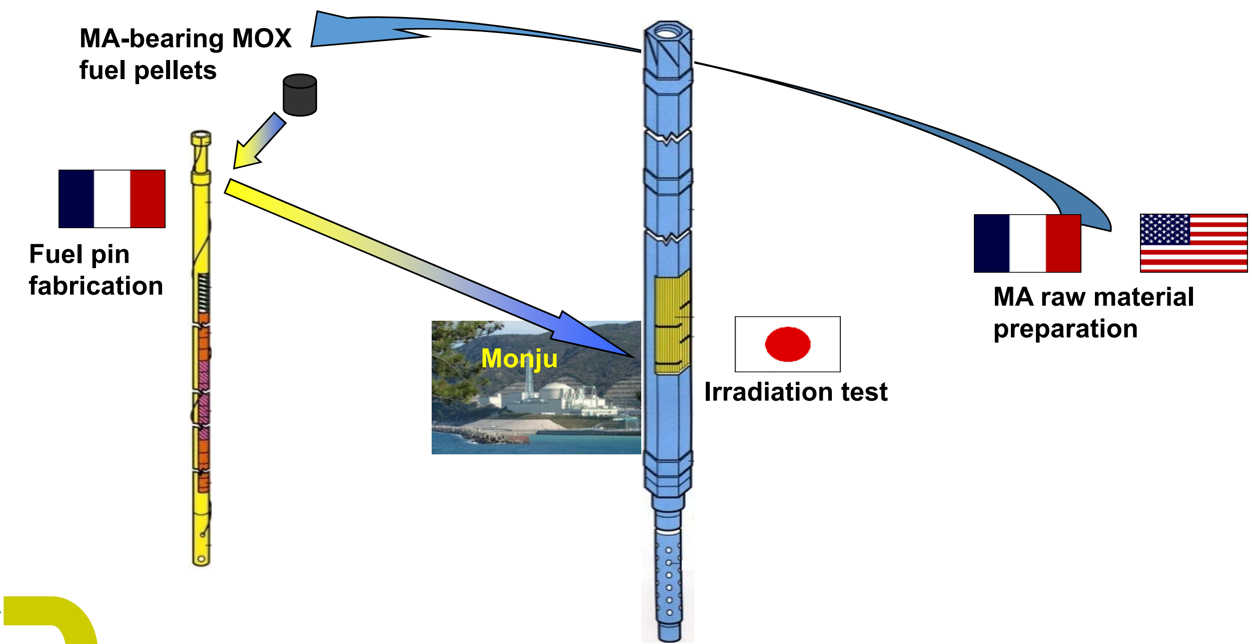
Advanced Fuel / Global Actinide Cycle International Demonstration / Component Design and Balance-Of-Plant / Safety and Operation / System Integration and Assessment X=Signatory, D=Under Discussion

Fuel Down Selection Report 2015

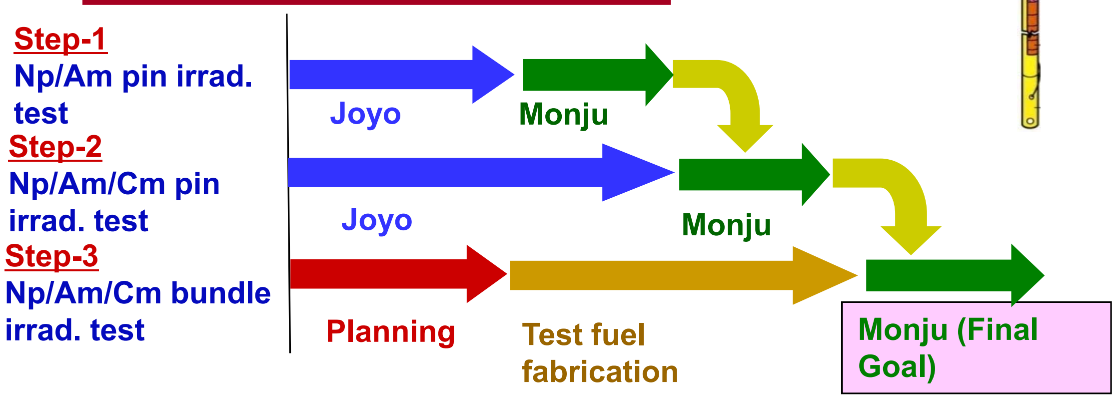
- The "Advanced Sodium Fast Reactor (SFR) Fuel Type Recommendation" was prepared by End of 2015
- **The final SFR fuel type selection for each member country is dependent upon multiple domestic factors**
- Oxide fuel is at the highest technical readiness level and has demonstrated adequate performance to high burnup at fuel pin level
- Metallic driver fuel has substantial experience demonstrated at the assembly level from the historical US program
- Nitride/carbide fuel have adequate performance demonstrated. However, the technologies are still in the early stages of assessment
- Target fuel systems dedicated to transmutation are also in the early stages of development and assessment
- ***China, France, Japan and Euratom select oxide fuel for initial SFR start-up. USA and ROK are working on metal fuel. Russia selected nitride fuel for BN-1200***
- Members recommend starting with ferritic/martensitic clad/wrappers but aim to transition in the longer term to other advanced alloys, such as oxide dispersion strengthened steels

Overview of GACID Project

- *A phased approach in three steps.*
- *Material properties and irradiation behavior are also studied and investigated.*



GACID overall schedule



- *The Project is being conducted by CEA, USDOE and JAEA as a GIF/SFR Project, covering the initial 10 years since Sep. 27, 2007.*

CD & BOP Project Subjects for 2012-2016

(1) In-Service Inspection & Instrumentation (ISI) technology

- Ultrasonic inspection in sodium using different approaches and technologies, codes and standards (CEA, Euratom, JAEA, KAERI)

(2) Repair experience

- Phénix, Monju, (CEA, JAEA)

(3) Leak Before Break (LBB) assessment technology

- Creep, fatigue, and creep-fatigue crack initiation & growth evaluation for Mod. 9Cr-1Mo (Grade 91) steel, Na leak detection by laser spectroscopy (JAEA, KAERI)

(4) Supercritical CO_2 Brayton Cycle Energy Conversion

- S-CO_2 compressor tests, cycle demonstration tests, compact heat exchanger tests, material oxidation tests, Sodium-CO_2 reaction tests, S-CO_2 plant dynamic analyses and control strategy development, S-CO_2 SFR design study, validation of S-CO_2 plant dynamic analyses with S-CO_2 loop data, sodium plugging tests (CEA, DOE, Euratom, JAEA, KAERI)

(5) Steam Generator design and associated safety & instrumentation (since 2011)

- Na/water reaction, thermal-hydraulics, thermal performance, DWT structural evaluation and heat exchange performance, DWT-SG fabrication (CEA, JAEA, KAERI)

Safety and Operations Project

Members

France (CEA)

US (USDOE)

Japan (JAEA)

Korea (KAERI)

EURATOM (JRC)

China (CIAE)

RF (Rosatom)

Project Objectives

- Analyses and experiments that support safety approaches and validate specific safety features
- Development and validation of computational tools useful for such studies
- Acquisition of reactor operation technology, as determined largely from experience and testing in operating SFR plants

Summary and Conclusions

- **Fast spectrum has favorable neutron balance which enables improved fuel cycle performance (resource utilization and waste management)**
- **The Sodium Fast Reactor (SFR) is the most mature of the Generation-IV technology options, with international demonstration experience**
- **Several collaborative Generation-IV R&D Projects are being conducted to explore technology innovations which target:**
  - **Improved economic performance**
  - **Robust behavior in off-normal conditions (inherent safety)**

QUESTIONS?

GEN IV International Forum[SM]

UPCOMING WEBINARS

25 January 2017	Very High Temperature Reactors	Mr. Carl Sink, DOE
22 February 2017	Gas Cooled Fast Reactors	Dr. Alfredo Vasile, CEA
28 March 2017	Supercritical Water Reactors	Dr. Laurence Leung, CNL