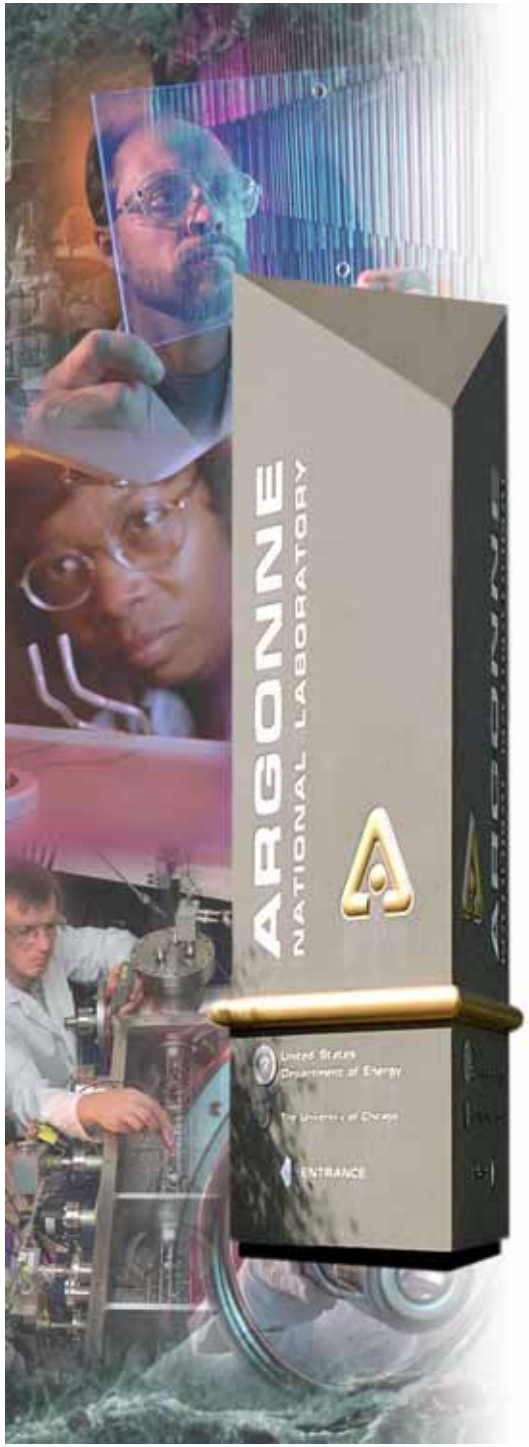

Reactor Design – Advanced Nuclear Systems

Workshop for Graduate Students from Big 10 Nuclear Engineering Departments
Argonne, Illinois
Wednesday, July 21, 2004

Temitope A. Taiwo
Nuclear Engineering Division

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Topics

Provide information on ANL activities in the definition and design of advanced nuclear reactors

- **Generation IV activities and systems**
- **Transmutation systems for nuclear waste minimization**
- **Backup information**
  - Additional design information
  - Technology and R&D needs for Gen IV systems

Generation IV

Nuclear energy systems deployable no later than 2030 in both developed and developing countries, for generation of electricity and other energy products

Generation I

Early Prototype Reactors

- Shippingport
- Dresden, Fermi I
- Magnox

Generation II

Commercial Power Reactors

- LWR-PWR, BWR
- CANDU
- VVER/RBMK

Generation III

Advanced LWRs

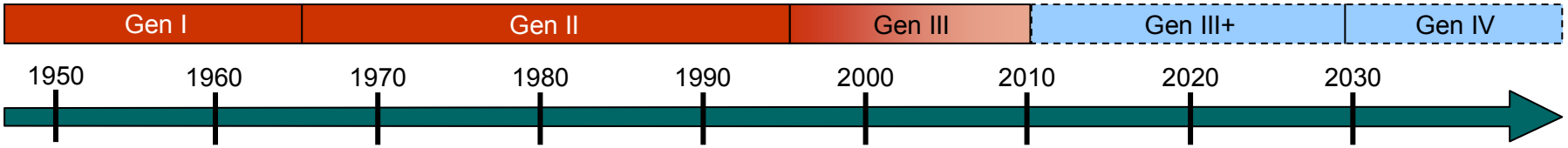
- ABWR
- System 80+
- AP600
- EPR

Near-Term Deployment

Generation III+ Evolutionary Designs Offering Improved Economics

Generation IV

- Highly Economical
- Enhanced Safety
- Minimal Waste
- Proliferation Resistant



Generation IV Systems

System	Neutron Spectrum	Fuel Cycle	Size	Applications	R&D
Very High Temp. Gas Reactor (VHTR)	Thermal	Open	Med	Electricity, Hydrogen, Process Heat	Fuels, Materials, H_2 production
Gas-Cooled Fast Reactor (GFR)	Fast	Closed	Med	Electricity, Hydrogen, AM	Fuels, Materials, Safety
Lead-alloy Fast Reactor (LFR)	Fast	Closed	Small to Large	Electricity, Hydrogen Production	Fuels, Materials compatibility
Sodium Fast Reactor (SFR)	Fast	Closed	Med to Large	Electricity, Actinide Mgmt. (AM)	Advanced Recycle
Supercritical Water Reactor (SCWR)	Thermal, Fast	Open, Closed	Large	Electricity	Materials, Safety
Molten Salt Reactor (MSR)	Thermal	Closed	Large	Electricity, Hydrogen, AM	Fuel, Fuel treatment, Materials, Safety and Reliability

Generation IV R&D Focus

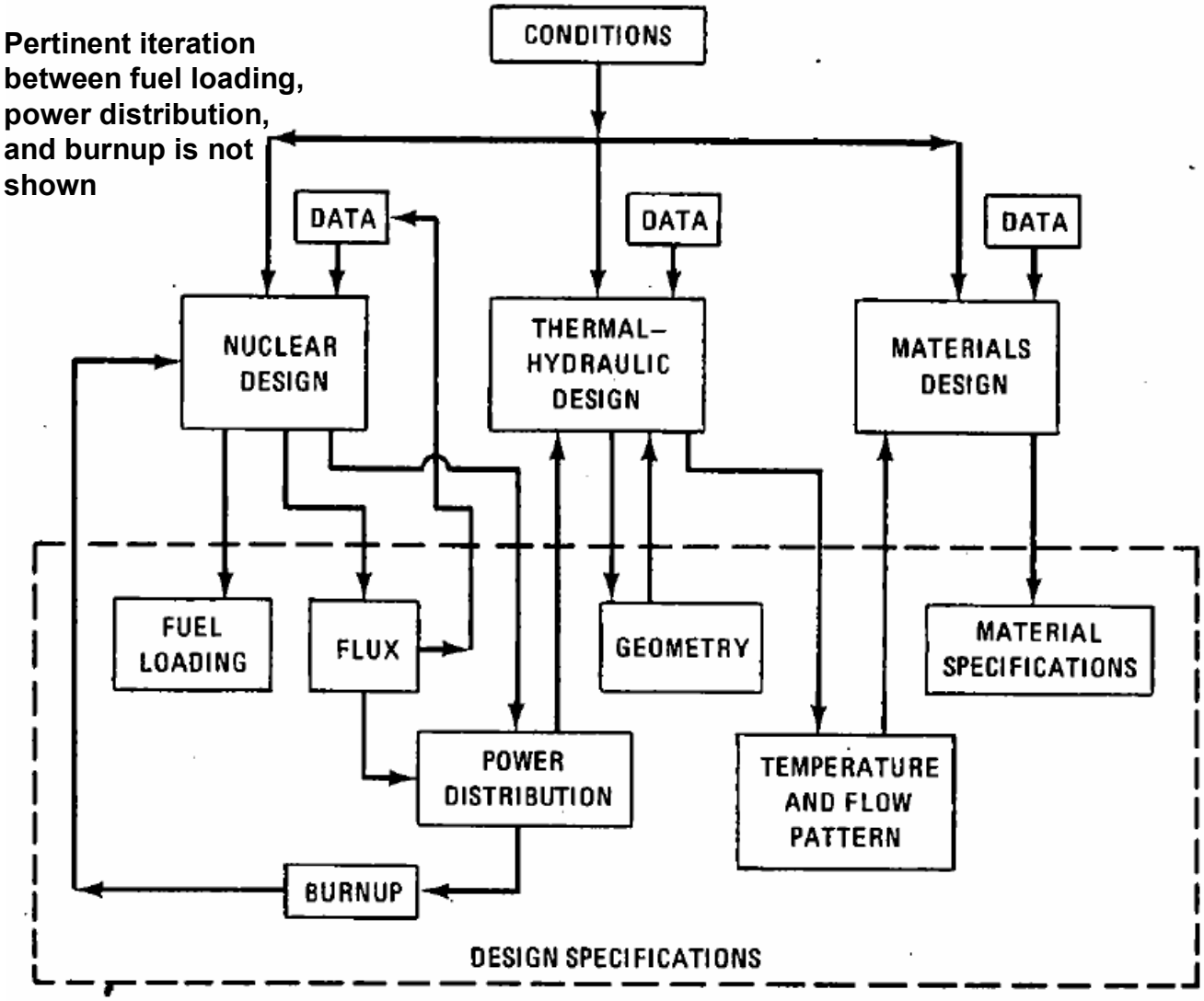
- **Thrust of Generation IV International Forum (GIF) collaborations is to develop and demonstrate advanced nuclear reactor systems**
  - vs. generic R&D
  - Steering committees formed for VHTR(NGNP), SCWR, GFR, and SFR
  - US participates in all, with NGNP receiving highest priority and funding
- **Early emphasis is on resolving key viability questions**
- **Conceptual design development is an integral part of the R&D**
  - Provide focus for technology development (fuels, materials, energy conversion, recycle)
  - Insure compatibility/integration of different technologies
  - Provide basis for evaluating performance

Range of System Characteristics

	VHTR	SCWR	GFR	LFR	SFR
Power, MW_{th}	600-800 (block) ~300 (pebble)	~2000-3600	600, 2400	25-400	800-3500
Power Density, W/cm^3	≤ 6.5	≤ 70	100 (50-200)	25-100	200-400
Primary Coolant (T_{Outlet}, ºC)	He (1000)	SC H_2O (450-500)	He (600-850) SC CO_2	Pb (500-800) Pb-Bi (500-550)	Na (510-550)
Fuel Material	UO_2, $UC_{0.5}O_{1.5}$	UO_2	(U,TRU) carbide, nitride, oxide	(U,TRU) nitride	(U,TRU) oxide, metal alloy
Fuel Form	Triso particle	solid pellet	CerCer dispersion, solid solution, coated particle	solid pellet	pellet or slug
Fuel Element/ Assembly	hex block, pebble	LWR or ACR type pin bundle	hex block, plate, pin, or particle	triangular pitch pin bundle	triangular pitch pin bundle w/duct
Moderator	graphite	water rods (PV) D_2O (PT)	None	None	None
Core Structural Material	graphite	F-M SS, Ni alloy	SiC matrix or cladding, TiN, ODS steel	F-M SS, SiC/SiC composite	ODS ferritic steel

Interactions between Different Disciplines

Pertinent iteration between fuel loading, power distribution, and burnup is not shown



- **Reactor designers determine set of system parameters that will give a safe, reliable, and economical reactor operation at rated power level over the core lifetime**
- **Entails interaction between different disciplines**
  - Reactor core physics; thermal-hydraulic (T-H) analysis of core cooling; structural analysis of core components; safety evaluation; and economic performance
  - Three primary analytical areas are shown: Nuclear, T-H and Materials

Common Issue – Passive Safety

Passively safe designs introduce additional issues

- **Safety features result in transients dominated by natural circulation and flows driven by small driving heads**
  - CFD-based codes may be required to resolve flow issues – codes should become more efficient and contain adequate core models
- **Systems are designed to minimize reactivity control requirements**
  - Small reactivity swings and the accuracy of codes have to be confirmed by experimental data
- **Reliable and effective thermo-structural reactivity feedback is key to the passive safety/passive load following design**
  - Strategy requires coordinated neutronics/thermalhydraulics/structural design of core

Reactor Physics Design Issues

System	Issues	
VHTR	•Fuel double heterogeneity •Stochastic behavior of pebble movement (for PBR variant)	•Graphite scattering treatment •Neutron streaming through coolant channels •Core/reflector interfacial effect
GFR	•Data for actinides, coolant (e.g., Pb, Bi) and fuel matrix candidate materials •Neutron streaming •Full-core transport effects •Spectral transition at core periphery •Modeling of reactivity feedback coefficients including expansion feedback	
LFR		
SFR		
SCWR	•Similar to BWRs •Increased heterogeneity	•Strong coupling of neutronics and T-H •Neutron streaming
MSR	•Evolution of mobile-fuel composition •Delayed neutron precursor loss	•Modeling of nuclear, thermal, and physio-chemical processes

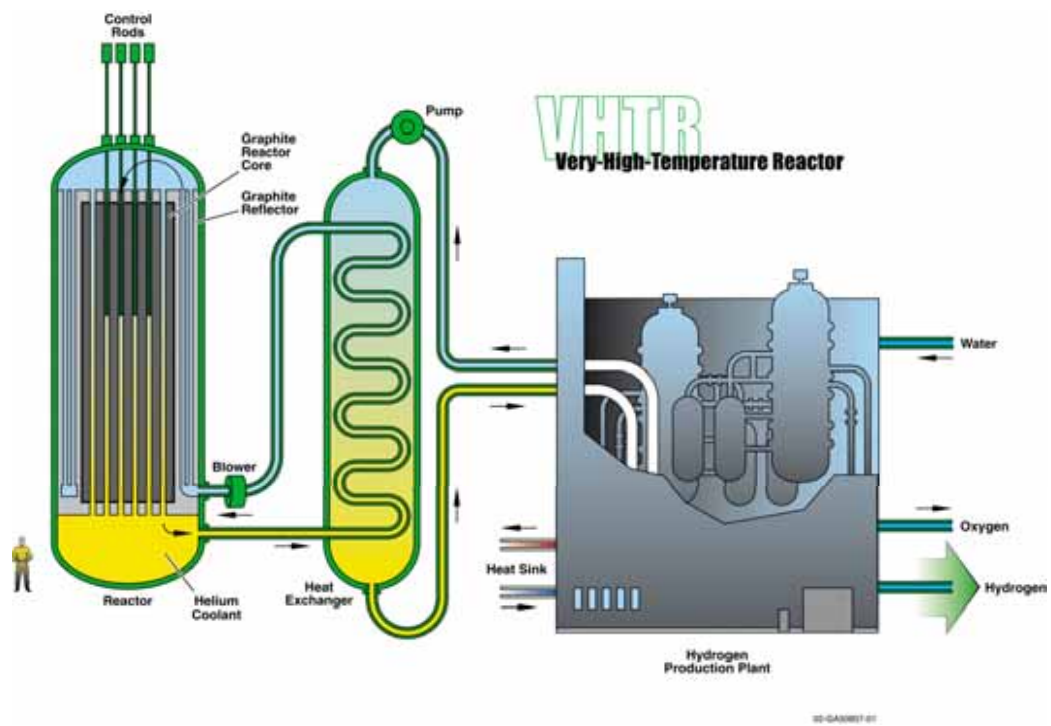
Very-High-Temperature Reactor (VHTR)

Characteristics

- He coolant, direct cycle
- 1000°C outlet temperature
- 600 MW_{th}, nominally based on GT-MHR
- Coated particle fuel – pebble or block fuel
- Solid graphite block core
- High thermal efficiency
- Hydrogen production
- Passive safety

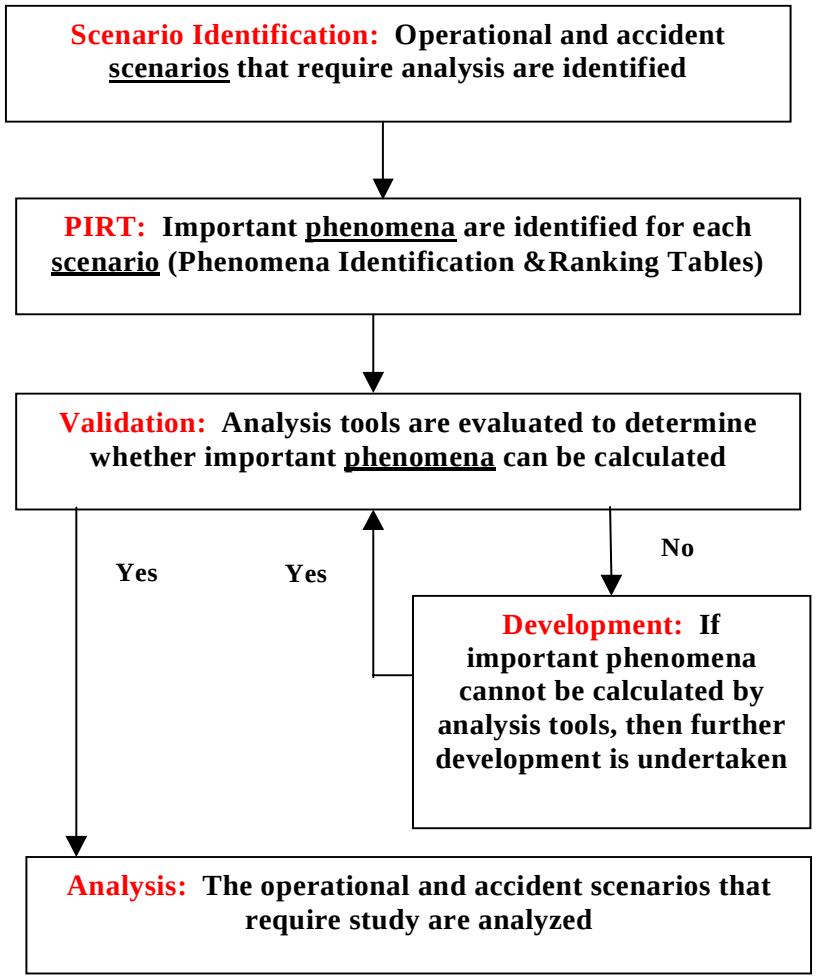
R&D

- Fuels for increased operating and transient temperatures
- High temperature materials development (metallic and ceramic materials for ex-core)
- Development of the reactor interface with the energy conversion system



Code Verification and Validation is a Major National Laboratory Role in NGNP (VHTR) Design

- **Collaborative activity involving ANL and INEEL in this area**
  - T-H/safety and physics aspects
- **It is unlikely that a national laboratory will be designer of the Next Generation Nuclear Plant (NGNP)**
- **Roles national laboratories may play within the USDOE NGNP program**
  - Development/improvement of design tools
  - Validation and verification of tools
  - Confirmation of design and safety
  - Technical support for plant licensing



Assessment of Experiments and Benchmark Tests – Focus on NGNP

- Identify and assess experiments and benchmark tests applicable to qualification and quality assurance of computer codes and databases for reactor physics analysis of NGNP
- Activity will support subsequent efforts (in FY 2005 and beyond) to document benchmark specifications and measured results in a standard format for use in NGNP software quality assurance

Approach

- ***Review preliminary design of NGNP cores***
- ***Identify experiments and benchmark tests***
- ***Assessment of experiments and data***
- ***International interactions***

Assessment of Experiments and Data

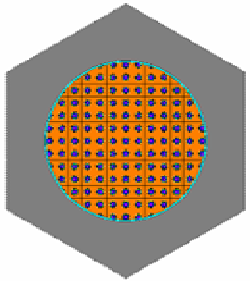
- Purpose of the experiment
- Descriptions of facility and experimental configurations -- geometry of core and fuel; core materials conditions
- Indication of physical parameters measured
- Are experimental uncertainties provided?
- Pedigree of experimental data
- Has a benchmark problem been developed from experimental data?
- Statement on applicability of data to V&V of methods for the NGNP
- Prioritization of cases assessed

Evaluation of Modeling Capabilities – Particle Heterogeneity and Distribution Effects

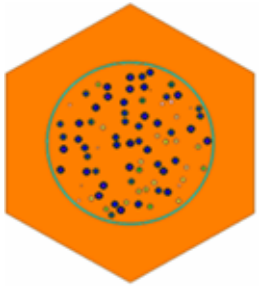
k_∞ Estimates:

			GT-MHR, UO_2	GT-MHR, $(TRU)O_{1.7}$	VHTR, $UC_{0.5}O_{1.5}$
MCNP4C	Random Distribution		**1.57335 ± 0.00040**	**1.25838 ± 0.00040**	**1.53280 ± 0.00082**
	Lattice Distribution	SC	1.57279 ± 0.00039	1.25427 ± 0.00071	1.52978 ± 0.00071
		BCC	1.57118 ± 0.00041	1.25317 ± 0.00070	1.53160 ± 0.00071
		FCC	1.57276 ± 0.00041	1.25192 ± 0.00077	1.52890 ± 0.00073
DRAGON			1.57565 (93)	1.26794 (599)	1.54393 (470)
WIMS8			1.57121 (-86)	1.25326 (-325)	1.52993 (-122)
Double heterogeneity effect			1.4 % $\Delta\rho$	13.1 % $\Delta\rho$	2.3 % $\Delta\rho$

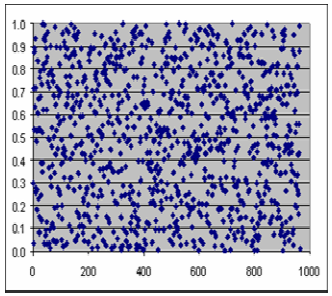
Lattice Distribution



Random Distribution (Planar)



Random Distribution (Axial)



Gas-Cooled Fast Reactor (GFR)

Nominal Characteristics

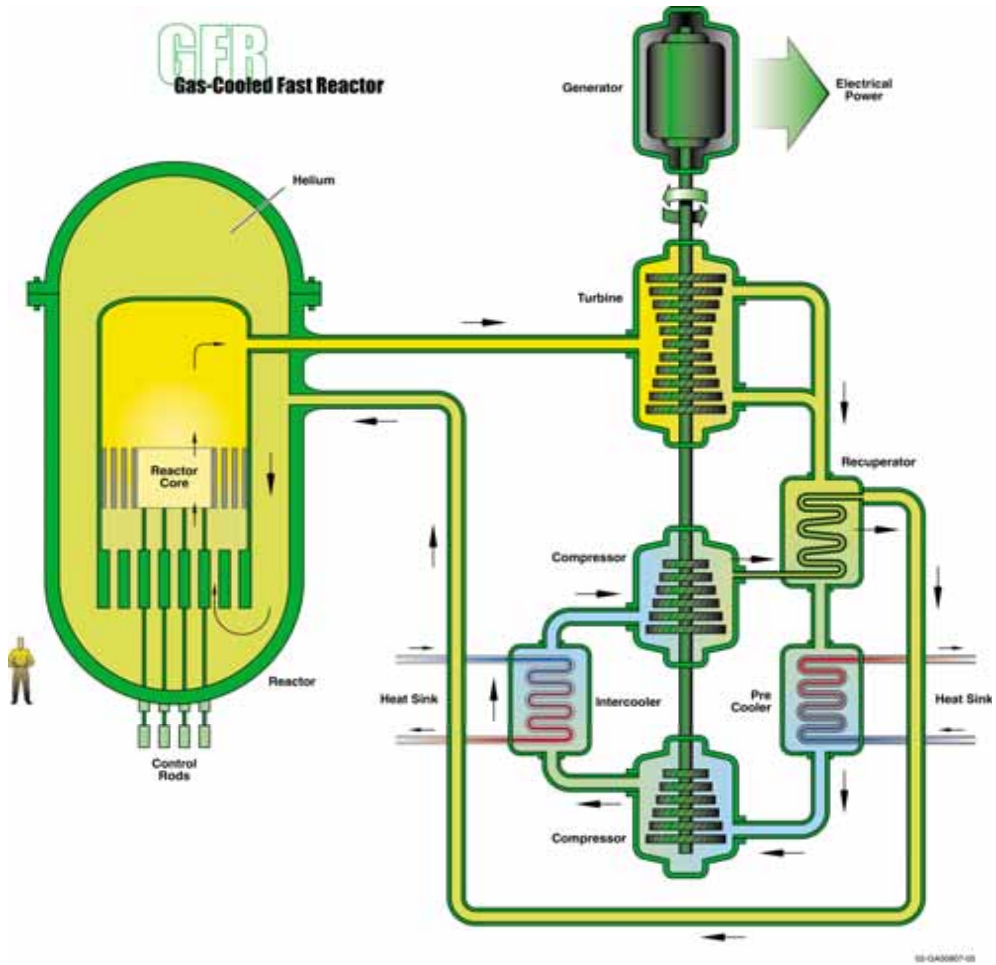
- He coolant, 850°C outlet temperature
- Direct cycle gas-turbine
- U-TRU ceramic fuel in CerCer dispersion, solid solution, or coated particle form
- Several fuel form and core geometry options

Benefits

- Efficient electricity generation
- Waste minimization
- High U resource utilization

Issues

- Fuel and materials development challenges
- Active safety systems needed at targeted power density due to relatively low thermal inertia and poor heat transfer characteristics of coolant at low pressure



GFR Design and Safety Work

- **Decay Heat Removal (DHR) strategy**
  - DHR could be realized by a **complete active system** (circulators, heat exchanger, valve); this system should be classified “safety”, redundant and diversified
  - A second system, for increased prevention of core degradation, could be designed/based on **natural circulation**, enhanced by a preliminary heavy gas injection phase (determined by pressure limitation in the containment building)
  - Design of the primary circuit and containment building should be adapted for sufficient back pressure (<15 bar) for a specified time
  - In addition, design provisions favoring conduction paths, energy storage, etc., should be investigated
- **Future work will optimize the current designs**

Lead-Cooled Fast Reactor (LFR)

Characteristics

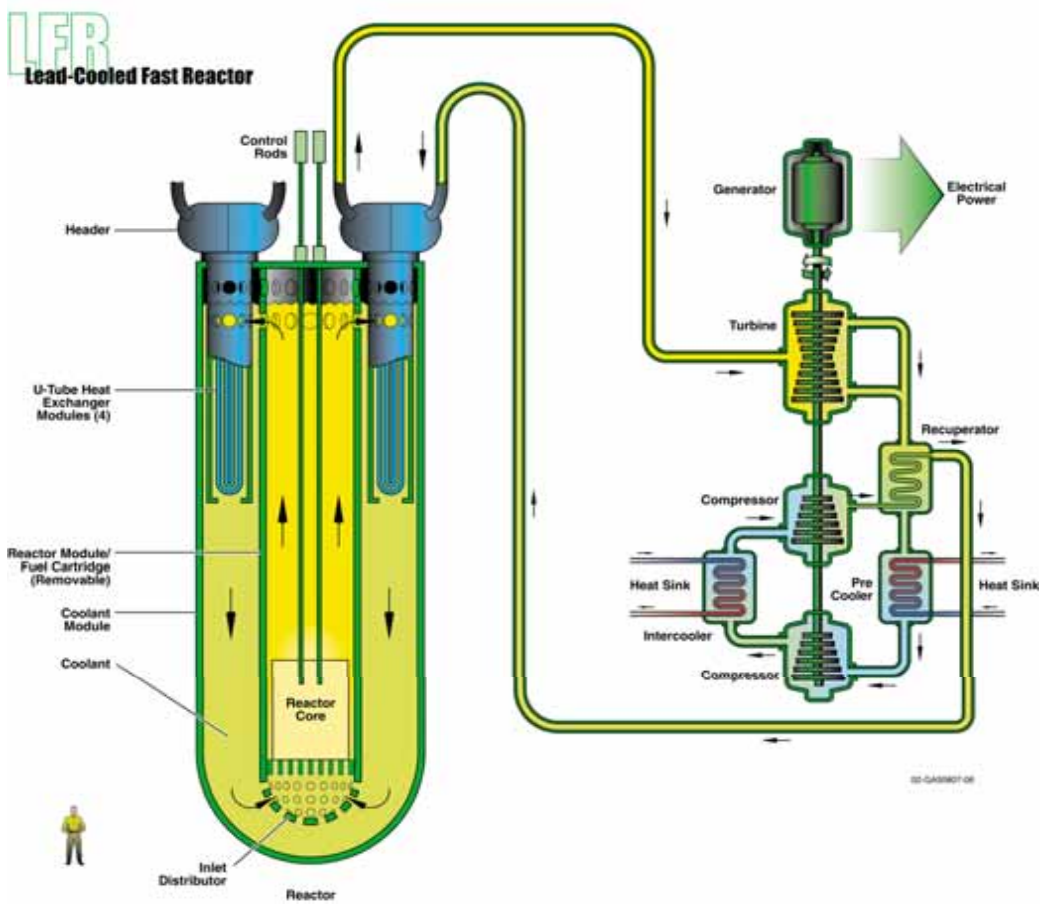
- 50–1200 MWe; Pb or Pb/Bi coolant
- 550°C to 800°C outlet temperature
- Fast Spectrum; multi-TRU recycle
- 15–30 year core life

Options

- Long-life (10-30 yrs), factory-fabricated *battery* (50-150 MWe) for smaller grids and developing countries
- Modular system rated at 300-400 MWe
- Large monolithic plant at ~1,200 MWe
- Long-term, Pb option is intended for hydrogen generation – outlet temperature in the 750-800°C range

Issues

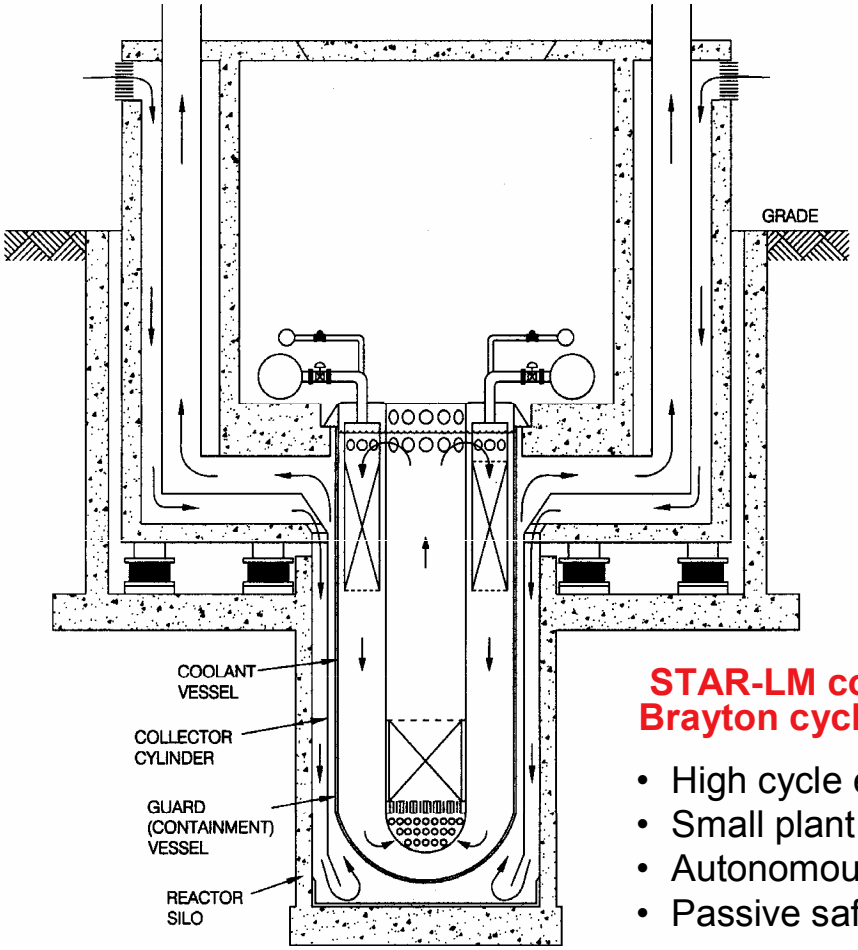
- Control of corrosion
- Coolant activation
- Seismic safety
- Qualification of Russian data



Key Features of Secure Transportable Autonomous Reactor (STAR-LM) LFR Coupled to S-CO_2 Brayton Cycle

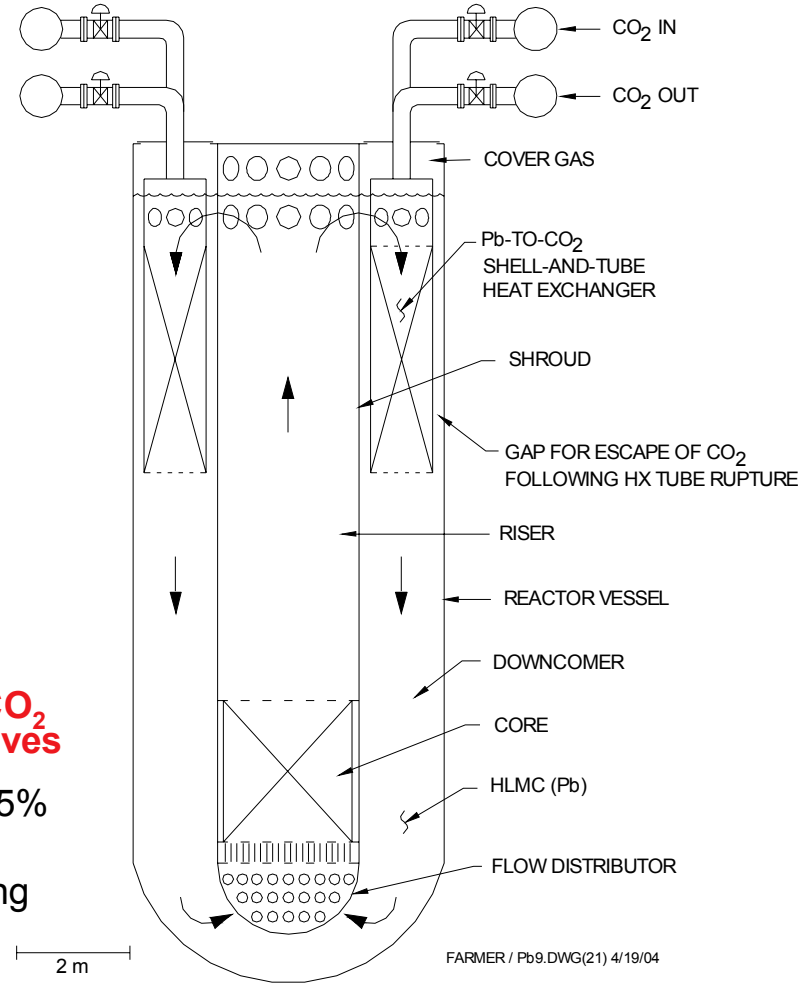
Power, MWe (MWt)	178 (400)
Primary Coolant	Lead (T_{melt} = 327 °C, T_{boil} = 1740 °C)
No Intermediate Coolant	Eliminated by inertness of lead that does not react chemically with CO_2 above 250 °C, and does not burn when exposed to air
Secondary Coolant/ Working Fluid	Supercritical carbon dioxide gas turbine Brayton cycle nominally at 20 to 7.4 MPa
Fuel Cycle	Sustainable closed fuel cycle, transuranic nitride (TRUN) cores are fissile self sufficient with depleted or natural U feed and minimal waste stream of fission products
Proliferation Resistance	Core is a single assembly/cassette that is not composed of individual removable assemblies and has an extremely long lifetime (15 to 20 years)
Autonomous Load Following	Core power adjusts itself to match heat removal from primary circuit by inherent physical processes due to strong reactivity feedback of fast spectrum core -- Coolant outlet temperature remains nearly unvarying with heat removal rate
Coolant Heat Transport	Natural circulation of heavy liquid metal at greater than 100 % nominal power --- Main coolant pumps eliminated

Illustration of STAR-LM Reactor

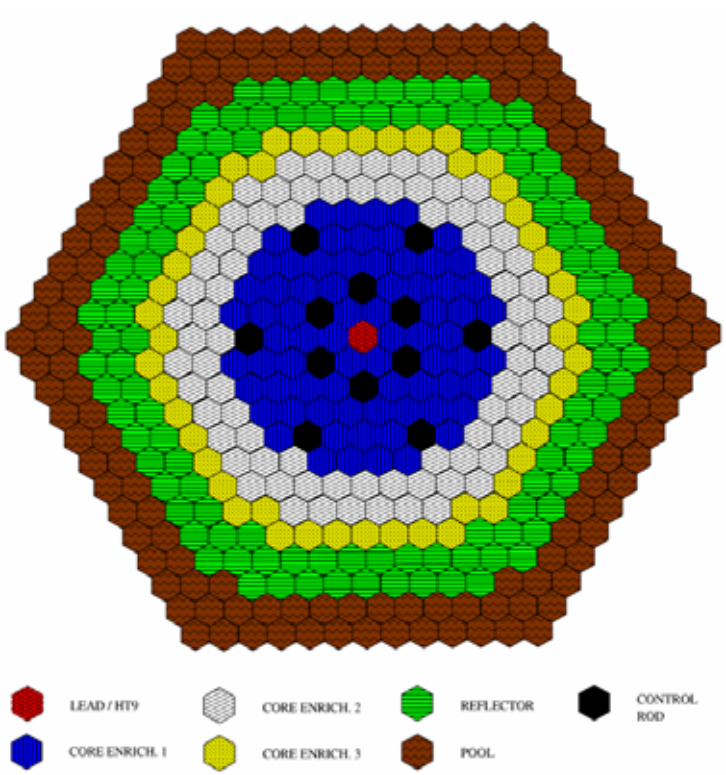


STAR-LM coupled to S-CO_2 Brayton cycle plant achieves

- High cycle efficiency of 45%
- Small plant footprint/size
- Autonomous load following
- Passive safety



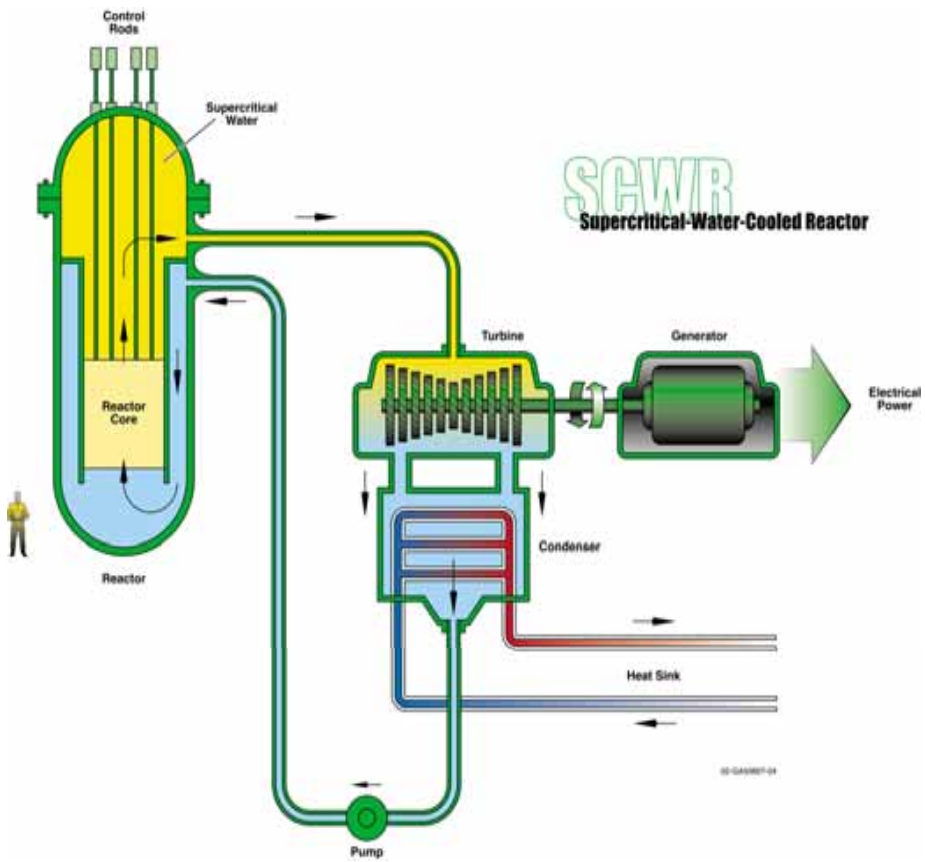
15 Full-Power-Year TRU STAR-LM Core Design Condition and Performance of STAR-LM Core



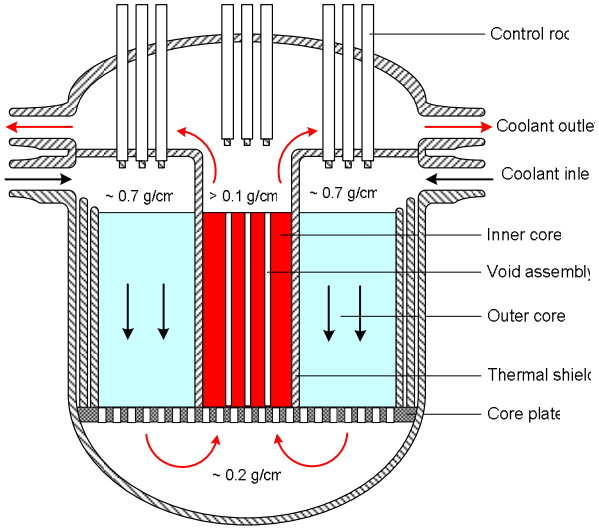
Power, MWt		400
Fuel Pin Diameter, cm		1.30
Fuel Pin Pitch-to-Diameter Ratio		1.536
Cladding Thickness, cm		0.1
Fuel Smeared Density, %		78.0
Volume Fraction	Fuel	0.2147
	Fuel-Cladding Bond	0.0606
	Cladding	0.0682
	Coolant	0.6566
Active Core Height, cm		200
Active Core Diameter, cm		246
Low Enrichment, %TRU/HM		13.30
Middle Enrichment, %TRU/HM		18.22
High Enrichment, %TRU/HM		21.28
BOC Loading, MT (tonnes)	HM	25.56
	U	21.47
	TRU	4.09
Core Power Density, W/cm^3	Average	43.58
	Peak	70.15
Peak Linear Heat Rate, W/cm		590.48
Discharge Burnup, MWd/kg	Average	83.11
	Peak	135.68
Peak Fast Neutron Fluence, 10^{23} n/cm^2		3.90
Peaking Factor (BOC/EOC)		1.63/1.64
Cycle Reactivity Swing, %Δk		0.61
Cycle Length, Full Power Years		15

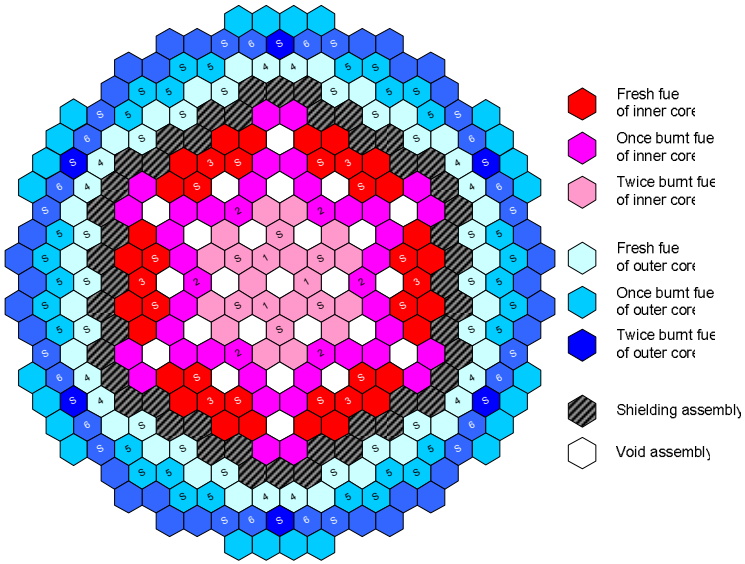
Supercritical Water (SCWR) Design

- **SCWR Characteristics**
  - Water coolant at supercritical conditions (~25 MPa)
  - Once-through direct cycle
  - 510°C outlet temperature
  - UO_2 fuel, clad with SS
  - Square assemblies with moderator rods
- **Large potential to improve economics**
  - Higher thermal efficiency
  - Compact plant systems
  - Low coolant mass flow rate per unit thermal power
- **R&D Items**
  - Testing of cladding and structural materials to demonstrate resistance to corrosion and strength under irradiation
  - Measurements to reduce uncertainty in SCW transport properties and correlations for heat transfer and fluid flow
  - Measurement of LOCA phenomena and adaptation/qualification of computer models
  - Verification of ability to prevent and control power-flow instabilities



Mixed-Spectrum SCWR ($MS^{2)}$ Concept



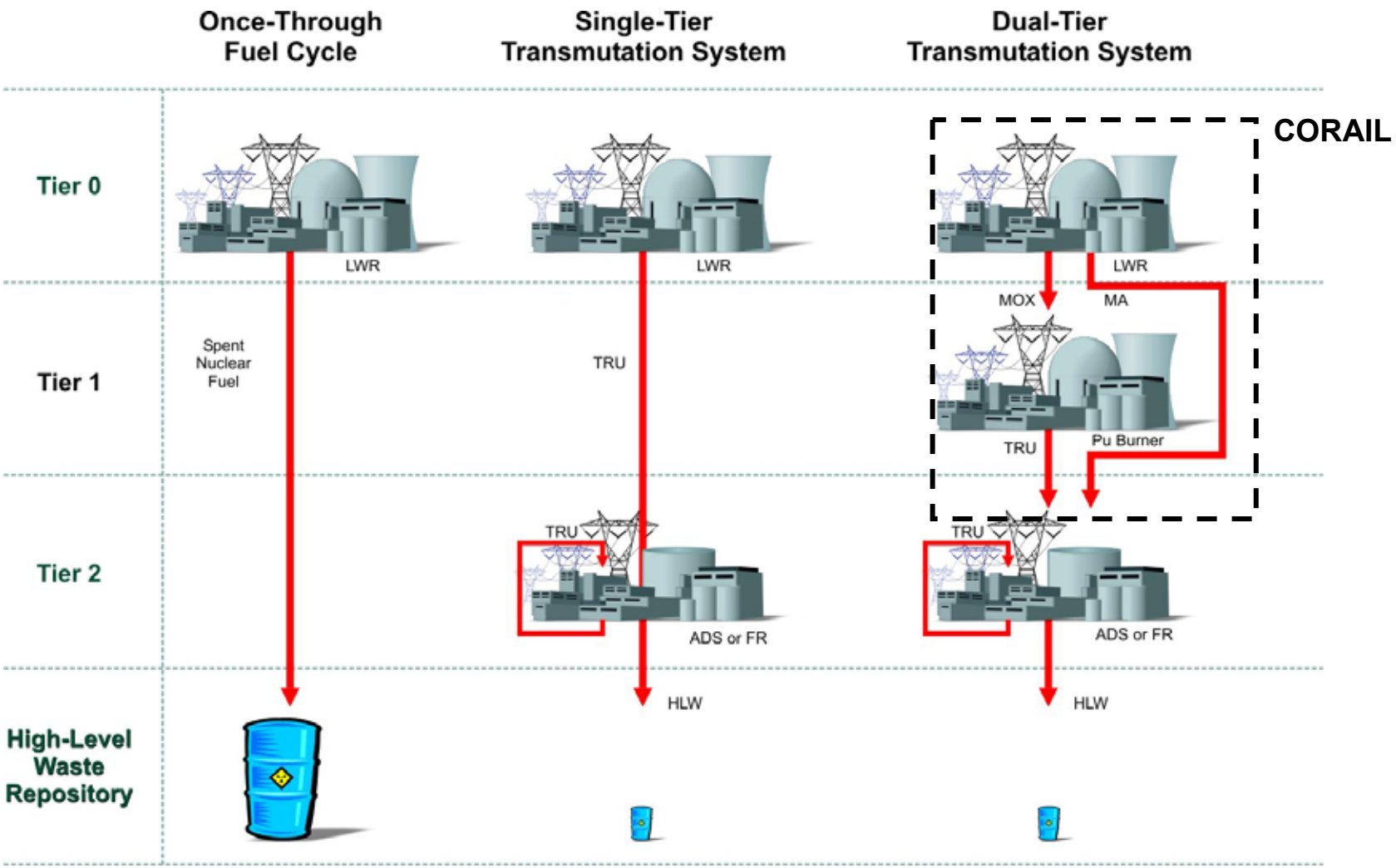


- **Advanced spectrum control is desirable to maximize merits of SCWR**
- **Mixed-spectrum supercritical water reactor**
  - Separation of fast and thermal spectrum zones
    - *Smaller power peaking factor and easier reactivity control*
    - *Sufficient control rod worth in thermal spectrum zone*
  - Multi-purpose reactor
    - *Maximize thermal efficiency and economy of SCWR concept without additional design features*
    - *Electricity production and actinide burning in a SCWR, simultaneously*
    - *High toxicity materials are confined in the fuel cycle (Net-Zero Pu, Np, and Am production)*

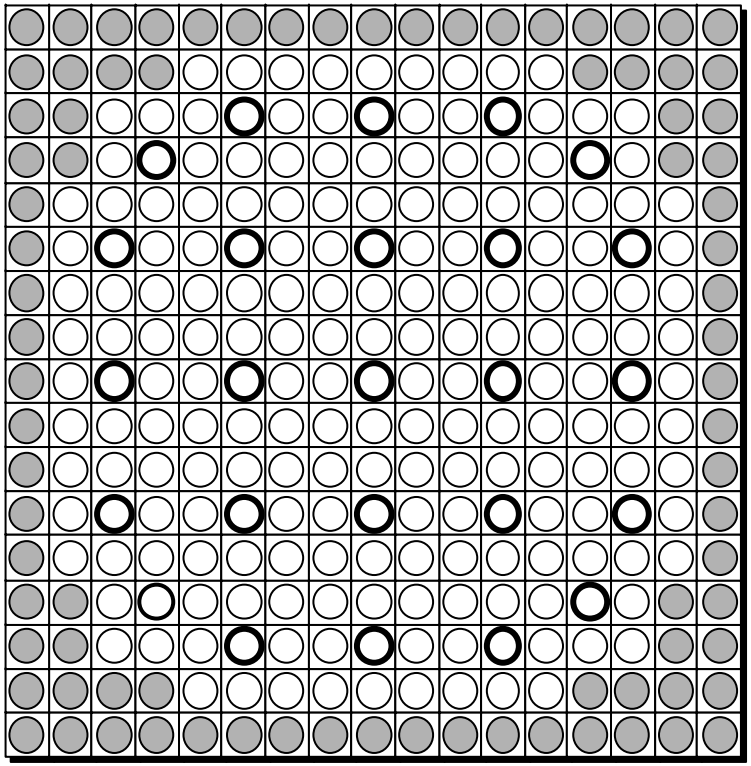
Comparison of Design Parameters

	Properties	Fast SCWR	Thermal SCWR	PWR	**MS² SCWR (Inner/Outer)**
General Core properties	Thermal power, MW	3893	3022	3411	**3400**
	System pressure, MPa	25	25	15.5	**25**
	Number of fuel assembly	278	121	193	**108/168**
	No. of shield/void assembly	-	-	-	**48/25**
	Active core height, m	3.20	4.27	3.76	**2.80**
	Effective radius, m	1.37	1.81	1.83	**1.25/2.03**
	Power density, MW/m^3	206.0	69.1	104.0	**156.7/84.8**
	Average linear heat rate, W/cm	221.02	194.97	178.00	**154.0/125.0**
Fuel and assembly	Fuel form	MOX	UO_2	UO_2	**TMOX/UO_2**
	Fissile enrichment, %	12	5	4	**~11/6.5**
	Cladding material	Ni-Alloy	ODS steel	Zr	**Ni-Alloy**
	Fuel radius, cm	0.4400	0.4470	0.4095	**0.4400/0.4095**
	Fuel pitch, cm	1.0100	1.1200	1.2500	**1.0000/1.2000**
	P/D ratio of fuel pin	1.14	1.25	1.52	**1.14/1.47**
	Assembly Shape	hexagonal	square	square	**hexagonal**
	Assembly pitch, cm	15.66	29.10	21.50	**20.71/20.71**
T/H	Inlet temperature, °C	280	280	300	**387/280**
	Average outlet temperature, °C	526	500	332	**553/387**
	Coolant mass flow rate, kg/sec	1694	1561	17222	**2000**

Transmutation System Approach



CORAIL Multi-Recycle Concept for Plutonium Stabilization



○ UO_2 rod ● MOX rod ◯ Guide tube

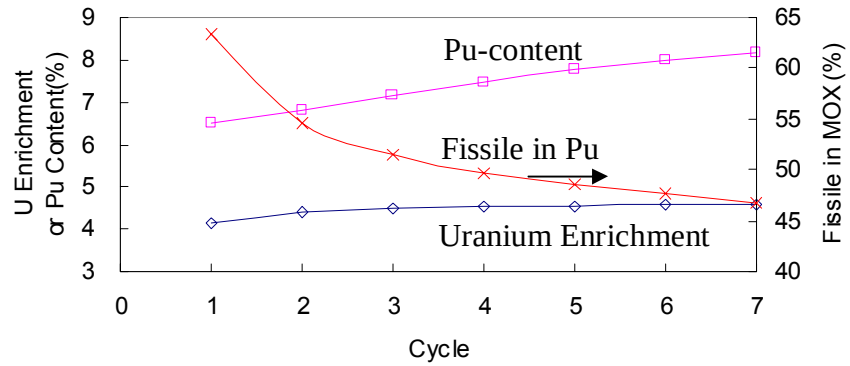
French-CEA CORAIL concept considered for Pu stabilization (i.e., no net production of Pu)

Compatible with existing LWR

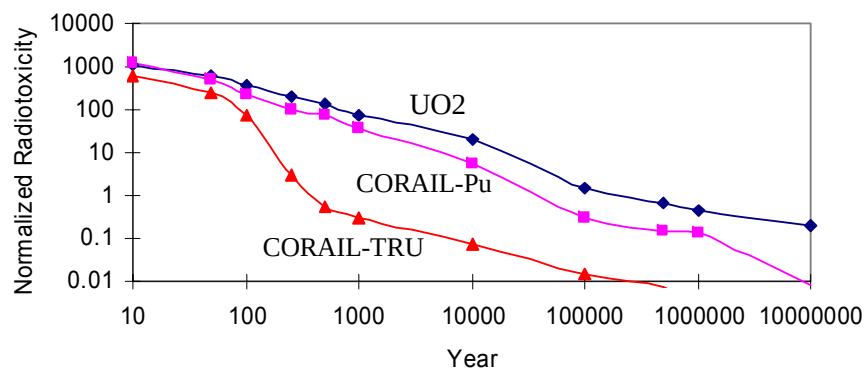
- **Concept**
  - **Heterogeneous assembly in a homogeneous core**
  - **Standard design using fuel rods and assembly that are qualified**
    - ***Mass balance in CORAIL core is similar to 30% MOX case, but much better for multirecycling***
  - **Pu/TRU discharged from both MOX and UOX pins is recycled**
- **Design Criteria**
  - **Uranium enrichment < 5.0%**
  - **Pu content in MOX < 12%**
  - **Power peaking factor < 1.2**
  - **No adverse effect on reactivity coefficients and shutdown margin**

Transmutation Performance of CORAIL-Pu Concept

- **No significant degradation of reactivity coefficients with multiple recycle**
- **30% MOX reaches equilibrium Pu isotopics within a few stages**
- **Solution required for minor actinides (MA)**
  - MA content higher than for UO_2 assembly (~3)
  - Direct disposal results in slight reduction of long-term radiotoxicity
  - Dual tier strategy sends minor actinide as fuel to Tier 2
- **Supporting studies pursued**
  - Detailed comparison of power distributions with CEA results



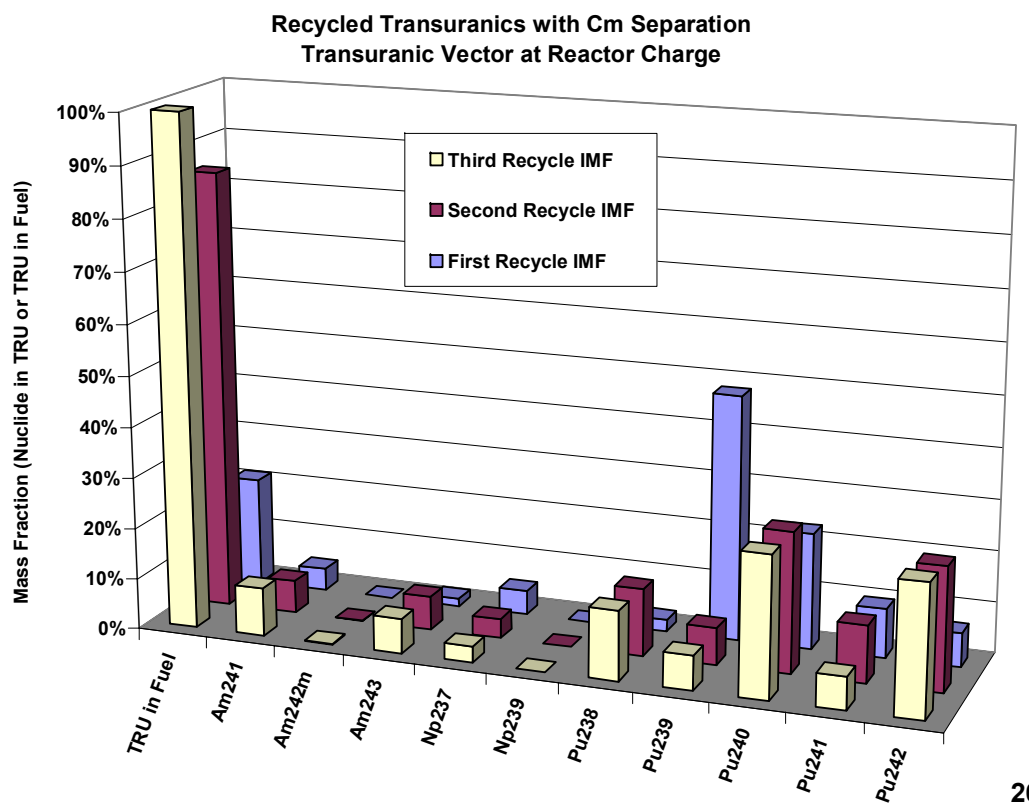
Mass Evolution with Recycling (CORAIL-Pu)



Normalized Cancer Dose

Inert Matrix Fuel (IMF)

- **Inert-matrix fuels (IMF) eliminate production of transuranics from U-238 conversion**
  - Very useful in weapons-grade Pu disposition program
- **Evaluated use of solid-solution $TRUO_2$-ZrO_2 for Pu+Np+Am recycling**
  - Assembly-level calculations over typical PWR fuel assembly residence time
  - Significant consumption of fissile-TRU and Am-241-chain nuclides in first recycle alone
    - *94% of Pu-239 consumed*
    - *48% of Am-241-chain consumed*
  - Further recycles perhaps not practical



Sodium-Cooled Fast Reactor – Concept Description

- **Sodium-cooled fast spectrum reactors using a closed fuel cycle with full actinide recycle**
- **Two major options:**
  - Intermediate size system (150-500 MWe) using U-Pu-MA-Zr metal fuel and a fuel cycle based on pyrometallurgical processing in co-located facilities
  - Medium to large size system (500-1500 MWe) with MOX (U,Pu) fuel supported by a fuel cycle based on advanced aqueous processing at a central location
- **Pool or compact loop primary system designs envisioned**

Fast Reactor Transmutation Analyses

Fast reactors with closed fuel cycle can effectively manage TRU

- **Can be configured as modest breeders (CR≥1) to moderate burners (CR≥0.5) with conventional technology**
- **Low conversion ratio designs (CR<0.5) have been investigated for transmutation applications in AFCI**
  - High enrichment fuels are required (~50% TRU/HM for CR=0.25)
  - Non-uranium fuel would be needed to achieve CR=0
- **Safety performance will change at low uranium content (e.g., reactivity losses, reduced Doppler coefficient)**
  - Detailed safety analysis conducted for CR=0.25 SFR system
  - Passive safety behavior is <u>not</u> compromised
- **Work has been initiated to develop fast burner designs for the other Generation-IV concepts**
  - Both gas-cooled (GFR) and lead-alloy cooled (LFR) concepts

Low Conversion Ratio SFR – Transmutation Performance

	High Leakage			Compact Core		
Targeted CR	**0.50**	**0.25**	**0.0**	**0.50**	**0.25**	**0.0**
Fuel Pin Diameter (cm)	0.74	0.64	0.49	0.67	0.59	0.45
Fuel Volume Fraction	0.38	0.26	0.14	0.30	0.22	0.12
Avg. Fuel Enrichment (%)	**30**	**51**	**90**	**31**	**51**	**91**
Heavy Metal Loading (MT)	14	8.4	4.1	7.5	4.6	2.5
TRU Loading (MT)	4.2	4.1	4.1	2.3	2.3	2.5
External Feed (MT/yr)	2.2	1.3	0.72	2.2	1.4	0.85
TRU consumption rate (kg/yr)	**126**	**198**	**270**	**122**	**193**	**259**
Burnup Reactivity Loss (%Δk)	2.8	4.6	6.4	2.6	4.3	5.9

- **Low conversion ratio designs developed for both high leakage (D=3.8 m) and compact (D=1.9 m) configurations**
- **Conventional enrichment at CR ~ 0.5**
- **Gradually increases to roughly 50% TRU/HM at CR ~ 0.25**
  - **Zr content increased to 40% for nonuranium case**
- **TRU Production Rate of LWR is ~250 kg/yr for 1 GWe**

Advanced Reactor Design Research Activities

- **In Generation IV Program, a wide variety of reactor technologies are being considered**
  - Each option has unique R&D focus items
  - Issues span a variety of disciplines
    - *Fuel development*
    - *Physics methods*
    - *Safety improvement*
    - *Nonproliferation and safeguards*
- **In AFCI Program, special reactor configurations and fuel types are being conceived to facilitate transuranic recycle**
  - LWR multi-recycle concepts
  - Burner fast spectrum concepts

BACKUP SLIDES

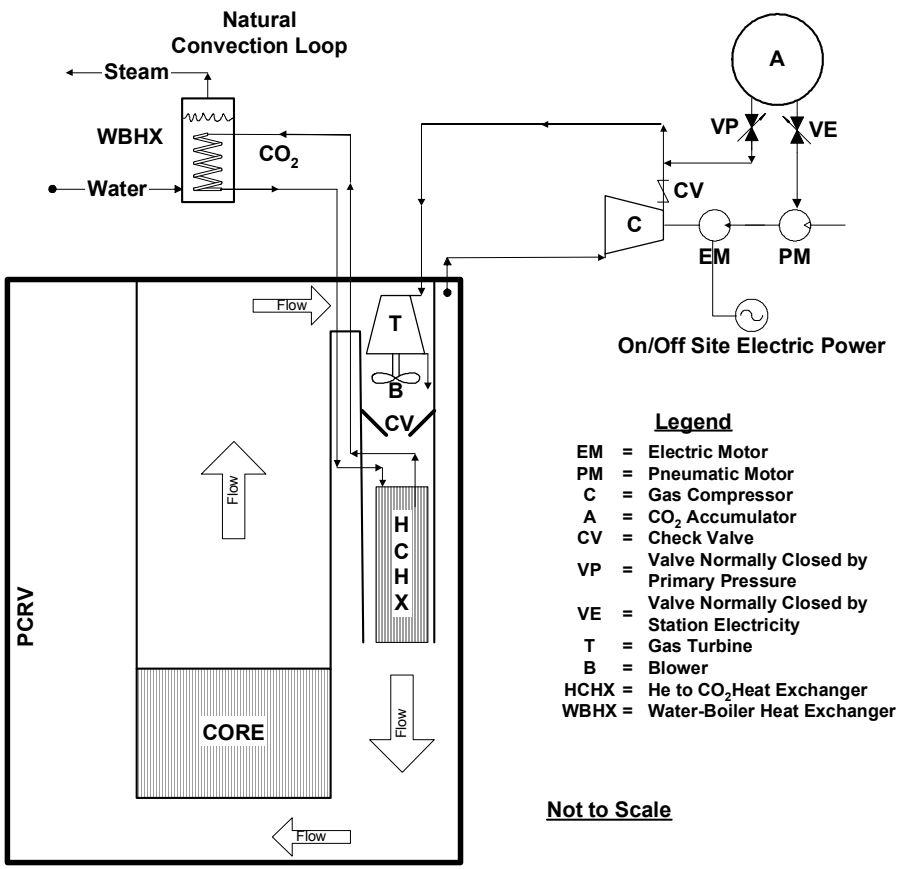
BACKUP SLIDES

Generation IV International Forum (GIF)

- **Ten nations are cooperating to advance Generation IV nuclear energy systems**
  - Argentina, Brazil, Canada, France, Japan, Korea, South Africa, Switzerland, UK, US
  - International organizations (Euratom, OECD-NEA, and IAEA) also participate
- **GIF Charter (July 2001)**
  - Establish and foster collaborations on future nuclear energy systems
  - Establish guidelines for collaborative R&D projects and reporting of their results
  - Regularly review progress and set future directions
- **Current GIF focus**
  - Agreements for multilateral collaborations
  - Joint R&D projects, based on Generation IV Roadmap

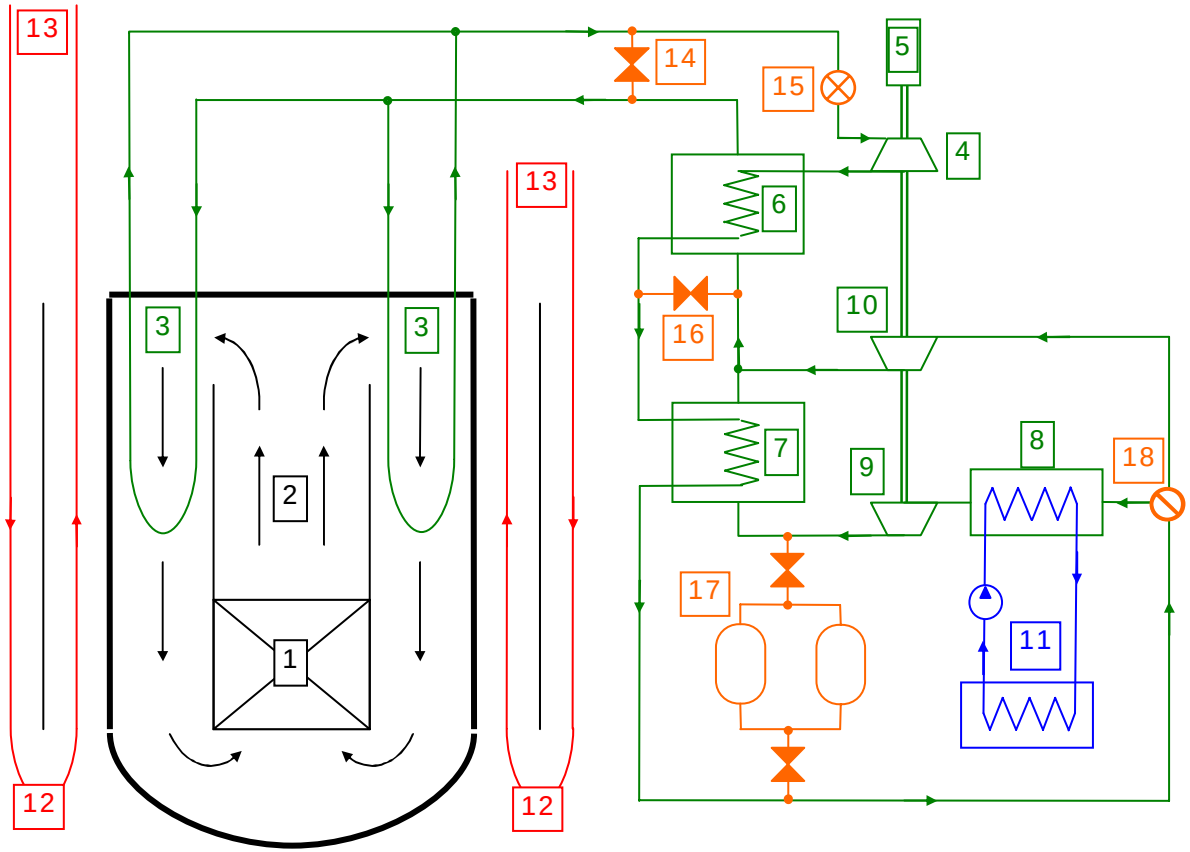
GFR Design and Safety Work

- **Safety systems capable of decay heat removal using natural circulation have been identified**
- **Heat storage, in core conduction, and vessel radiation**
  - Combination of these three alone could not realize an effective DHR approach
- **In core heat exchangers (heat pipes, cold fingers)**
  - Potentially very effective systems but have neutronic impact, technological difficulties, question of the ultimate heat sink, reliability, and safety impact
- **Forced circulation**
  - Very efficient : 3% nominal flow enables the core cooling while fulfilling fuel temperature criteria
  - Circulators of a very limited power (100 KW) meet the requirements



Pneumatically-powered, decay heat removal system

Schematic of STAR-LM Coupled to S-CO_2 Brayton Cycle Showing Heat Transfer Paths



1 – Reactor core
2 – Pb primary coolant (natural circulation)
3 – Pb-to-CO_2 in-reactor heat exchanger
4 – CO_2 turbine
5 – Generator
6,7 – High and low temperature recuperators
8 – Cooler
9,10 – Compressors
11 – Cooling circuit to ultimate heat sink or desalination plant
12 – Guard vessel natural circulation air cooling system
13 – Atmosphere heat sink
14 – In-reactor heat exchanger bypass valve
15 – Turbine inlet valve
16 – Turbine bypass valve
17 – Inventory control
18 – Flow split valve

Color coding:
Black – Primary circuit
Green – Normal heat removal
Blue – Heat sink
Red – Shutdown and decay heat cooling
Orange – Control system components

Key Features of STAR-LM/Brayton Cycle (Cont'd)

Passive Safety	Core remains covered by ambient pressure single-phase coolant which removes core power by natural circulation during all operational transients and postulated accidents
Decay Heat Removal	Vessel cooling by natural circulation of air; always in effect
Peak Cladding Inner Surface Temperature, °C	650
Pb Coolant Outlet Temperature, °C	588
Pb Coolant Inlet Temperature, °C	438
Peak Fuel Centerline Temperature, °C	708
CO_2 Temperature Exiting HX, °C	564
CO_2 Temperature Entering HX, °C	408
Brayton Cycle Efficiency, %	44.9

Key Features of STAR-LM/Brayton Cycle (Cont'd)

Seismic Isolation	**Nuclear island supported on three-dimensional seismic isolators to accommodate seismic and sloshing loads**
Physical Isolation	**Autonomous load following and passive safety enhance resistance to sabotage or malevolent human-induced acts. Reactor is situated below grade inside of a silo providing protection against an aircraft crash. Containment and building above vessel can be buried under dirt or other material**
Corrosion Control	**Formation of protective oxide films upon ferritic-martensitic stainless steel structure and maintenance of films without formation of solid lead oxide contaminants through control of dissolved oxygen concentration in primary coolant. Possible use of stainless steel coated with protective alumina layer. Possible alternative corrosion control schemes without need for oxygen control such as formation of surface layer that is resistant to attack**

Sodium LMR R&D

- **Engineering development of advanced aqueous technology**
  - Powdering technology for fuel de-cladding
  - Crystallization method for extraction of excess U
  - Co-extraction of U/Pu/Np; low-waste recovery of Am and Cm
  - Simplified pellet fabrication process
- **Development and scale-up of pyroprocess technologies**
  - Head-end reduction of LWR spent fuel to metal
  - Recovery of transuranics from metallic fuel
  - Waste reduction and waste-form qualification
- **Fuel development**
  - Remote fabrication of minor actinide bearing fuels
  - Performance of recycled fuel
- **Improvement of capabilities for in-service inspection and repair**

VHTR R&D

- **Fuels for increased operating and transient temperatures**
  - High temperature coatings (e.g., ZrC)
  - Fuel design for reduced temperature rise
- **High temperature materials development**
  - Metallic materials for IHX, gas ducts/pipes, isolation valves (e.g., Ni-Cr-W super-alloy)
  - Ceramic materials for T >950°C (e.g., C/C composites, super-plastic ceramics)
- **Development of the reactor interface with the energy conversion system**

SCWR R&D

- **Testing of cladding and structural materials to demonstrate**
  - Resistance to corrosion and stress corrosion cracking
    - *For temperatures up to 620°C*
    - *For radiation doses up to 30 dpa (thermal); 150 dpa (fast)*
    - *Accounting for water chemistry and radiolytic decomposition*
  - Dimensional and micro-structural stability
  - Strength, ductility and creep-resistance as a function of irradiation dose and temperature
- **Safety R&D**
  - Measurements to reduce uncertainty in SCW transport properties and correlations for heat transfer and fluid flow
  - Measurement of LOCA phenomena and adaptation/qualification of computer models
  - Verification of ability to prevent and control power-flow instabilities

Pb/Bi Battery R&D

- **Development/verification of cladding and structural materials**
  - Fabricability, compatibility with fuel and coolant, resistance to irradiation damage, retention of strength and ductility – over 15 to 20 year service life
  - Monitoring and control of coolant chemistry
- **(U,TRU)nitride fuel and recycle development, building on existing programs in Japan and Europe**
- **Development of T/H data base for natural circulation heat removal**
- **Development of energy conversion technologies**
  - Heat exchangers to transport heat to working fluid or hydrogen plant
  - Supercritical CO_2 turbine
  - Calcium-Bromine process for hydrogen production

GFR R&D

- **Screening and evaluation of high actinide-density fuels**
  - Modified coated particle or dispersion type fuels, e.g.,
    - *(U,TRU)C/SiC*
    - *(U,TRU)N/TiN*
  - Fuel pins with high-temperature cladding
- **Core structural materials for high-temperature and fast fluence conditions (ceramics, composites, refractory alloys)**
- **Core configuration for enhanced thermal inertia and passive dissipation of decay heat**
- **Fuel recycle technologies**
  - Separation of fuel compound from matrix
  - Aqueous and dry recycle options; ^{15}N recovery from nitride

MSR R&D

- **Selection/characterization of fuel salt for actinide consumption application**
  - Limit generation of tritium
  - Verify solubility of minor actinides
  - Characterize behavior of fission and activation products
  - Determine thermo-physical data
- **Develop processing technology for fuel salt leading to waste forms of acceptable purity and durability**
- **Demonstrate performance capabilities of metallic and graphite structural materials**
  - Lifetime compatibility with molten salt constituents
  - Durability in radiation environment
- **Develop technical means for resolving such challenges as**
  - Preventing noble metal fission product plate-out on walls of the intermediate heat exchanger
  - Limiting tritium diffusion from primary system

Crosscut R&D Recommendations

Crosscut Area	Research Topic	Na LMR	Pb/Bi Battery	GFR	VHTR	SCWR	MSR
Fuel Cycle	Integrated once-through fuel cycle				X	X(T)	
	Optimum management of Cs and Sr	X	X	X		X(F)	X
	Cm management and target fabrication	X(O)	X	X		X(F)	
	Aqueous processing with group separation of Actinides	X(O)	X	X		X(F)	
	Pyroprocess actinide recovery optimization	X(M)	X	X			

M: Metal Fuel
O: Oxide Fuel

T: Thermal Spectrum
F: Fast Spectrum

Crosscut R&D Recommendations, cont'd

Crosscut Area	Research Topic	Na LMR	Pb/Bi Battery	GFR	VHTR	SCWR	MSR
Fuels and Materials	Properties and behavior of structural materials						
	T < 600°C	X	X			X	X
	600°C < T < 900°C		X	X		X	X
	900°C < T			X	X		
	Development of fuel fabrication techniques	X	X	X	X	X	
	Irradiation and transient testing campaigns	X	X	X	X	X	X
	Fundamental modeling of materials behavior	X	X	X	X	X	X
Energy Products	Analysis of market requirements		X	X	X		X
	Thermochemical water splitting processes		X	X	X		X

Crosscut R&D Recommendations, cont'd

Crosscut Area	Research Topic	Na LMR	Pb/Bi Battery	GFR	VHTR	SCWR	MSR
Economics	Modular fabrication and installation technologies	X	X	X	X		
	Instrumentation, control, and human-machine interface	X	X	X	X	X	X
Risk and Safety	Radionuclide transport and dose assessments	X	X	X	X	X	X
Evaluation Methods	Economics: consistent, integrated cost and revenue models	X	X	X	X	X	X
	Enhanced probabilistic risk assessment tools	X	X	X	X	X	X
	Proliferation resistance and physical protection criteria and methods	X	X	X	X	X	X