The Advanced High-Temperature Reactor

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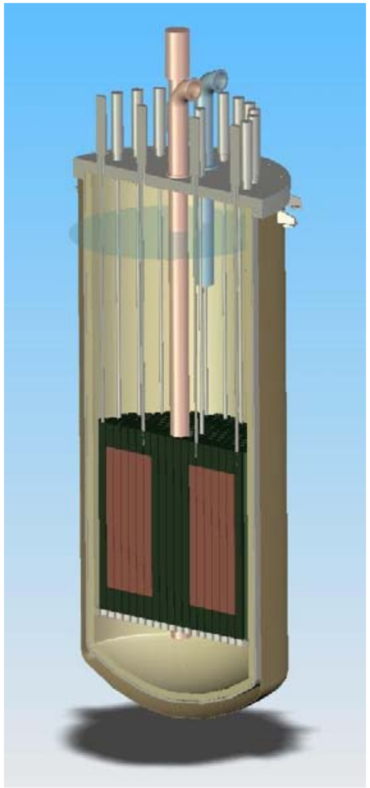

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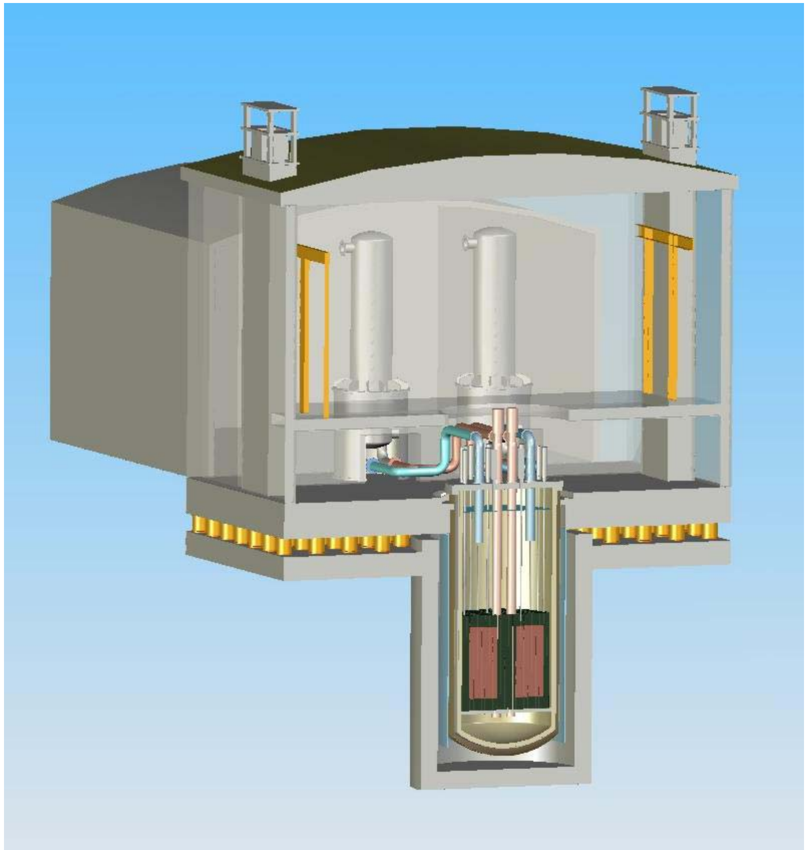
OAK RIDGE NATIONAL LABORATORY

Sandia National Laboratories

UC Berkeley NUCLEAR ENGINEERING

Contents



- Introduction
- Fuels
- Molten Fluoride Salt Coolants
- Materials
- Facility Design
- Reactor Core Design
- Energy Conversion
- Economics
- Research And Development

History, Characteristics, and Status: The Advanced High-Temperature Reactor

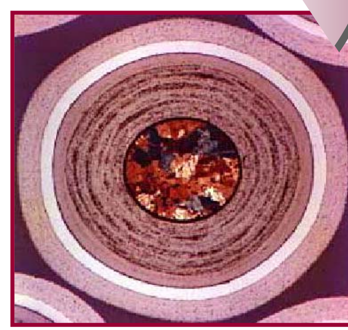
- New reactor concept that combines three characteristics in a single reactor
  - High temperature for electricity and H_2 production
  - Passive safety (same safety basis as modular gas-cooled reactors)
  - Large power output (improved economics)
- Joint effort
  - Oak Ridge National Laboratory
  - Sandia National Laboratories
  - University of California at Berkeley
- A series of studies and evaluations have been conducted, but a point design has not not yet been fully developed

The Advanced High-Temperature Reactor

Combining Existing Technologies In A New Way

Passively Safe Pool-Type Reactor Designs

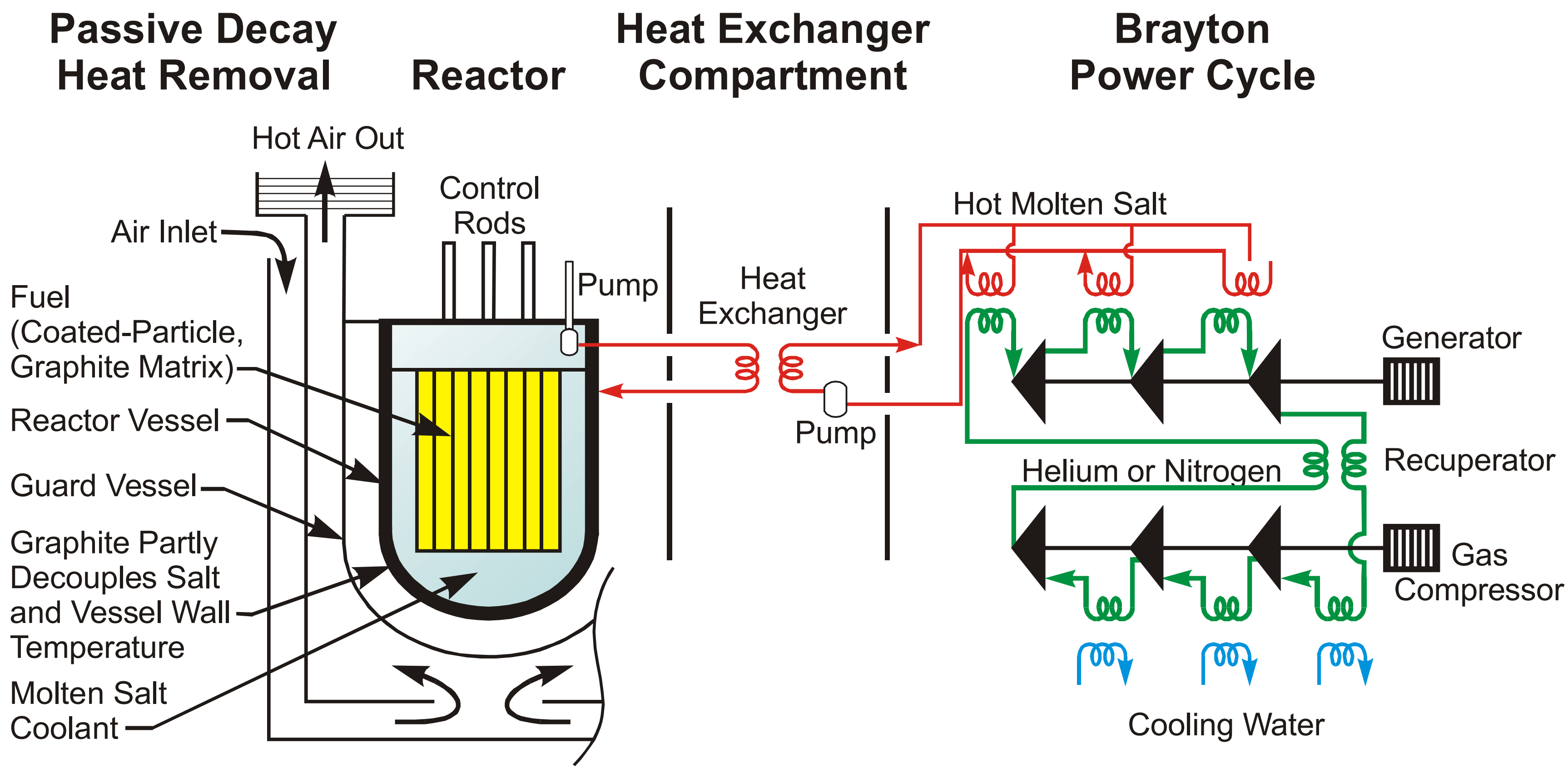
Brayton Power Cycles



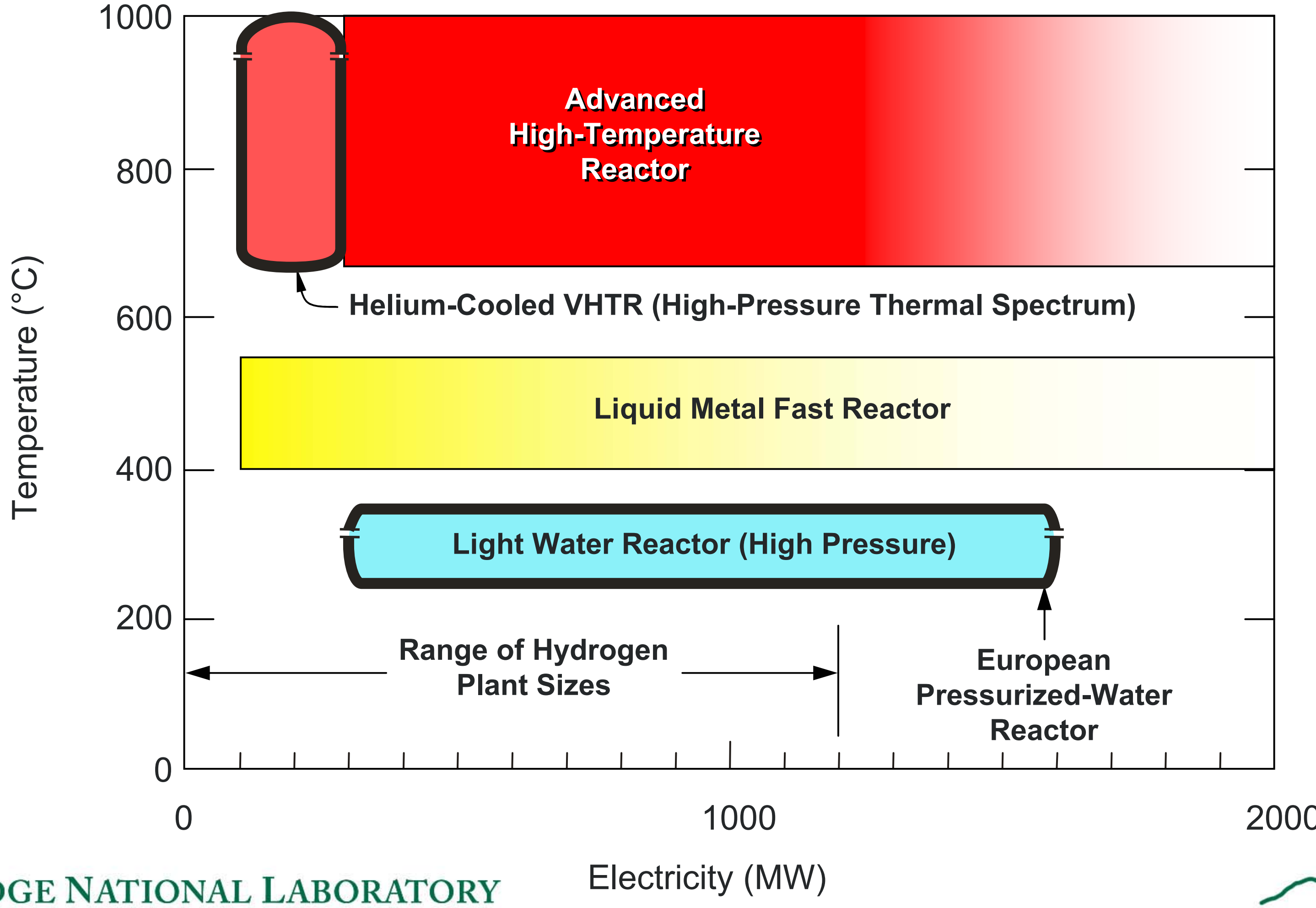
High-Temperature Coated-Particle Fuel

High-Temperature, Low-Pressure Transparent Molten-Salt Coolant

The Advanced High Temperature Reactor



The AHTR Extends High-Temperature Capabilities To Large Reactors

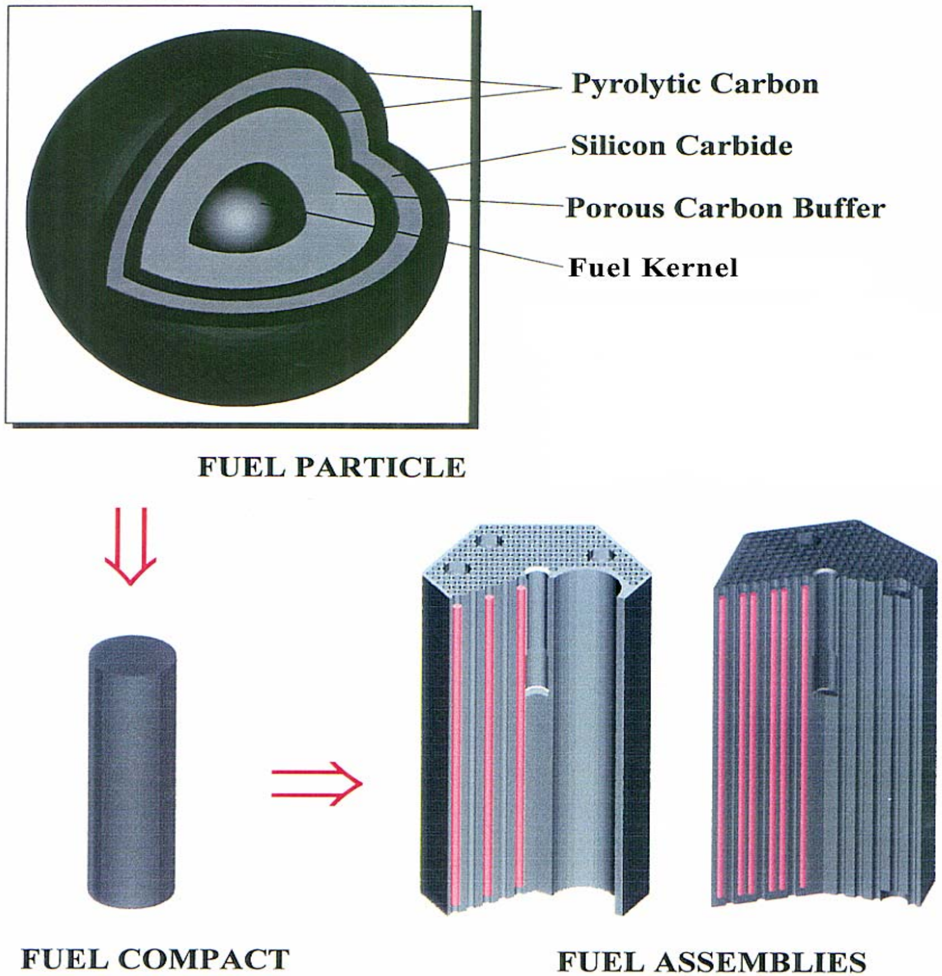


Fuels

Only One Type of High-Temperature Nuclear Fuel Has Been Demonstrated on a Significant Scale

The AHTR Uses Coated-Particle Graphite Fuel Elements
(Peak Operating Temperature: 1250ºC; Failure Temperature >1600ºC)

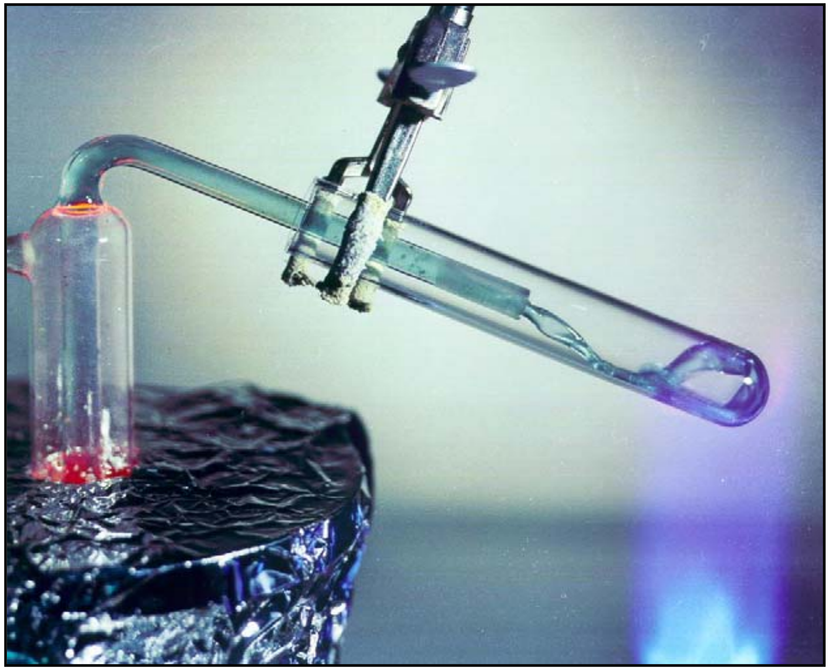
Same Fuel as Used in Gas-Cooled Reactors



- Fuel particle with multiple coatings to retain fission products
- Fuel compact contains particles
- Compacts inserted into graphite blocks
  - Several options for graphite geometry (prismatic, rod, pebble bed, etc.)
  - Base design uses prismatic; other options viable
- Graphite block supports fuel compacts in an arrangement compatible with nuclear reactions and heat transfer to coolant

Molten Fluoride Salt Coolants

Only Two Coolants Have Been Demonstrated As Compatible with High-Temperature Graphite Fuels



Helium

(High-Pressure/Transparent)

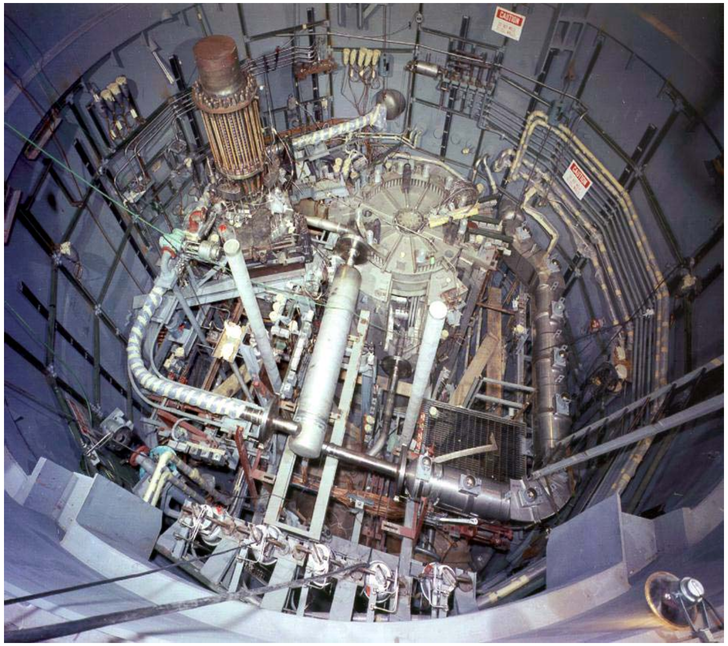
Molten Fluoride Salts

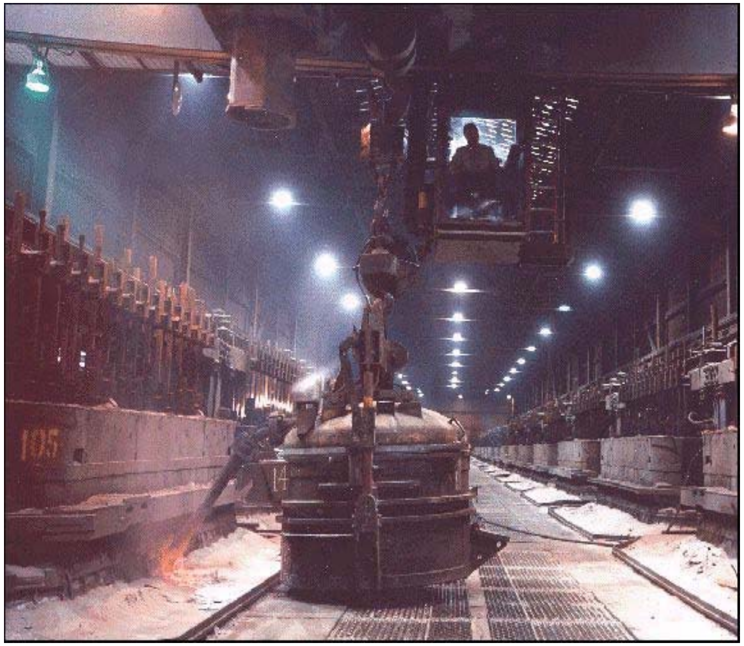
(Low Pressure/Transparent)

The AHTR Uses a *Molten Salt Coolant*

Good Heat Transfer, Low-Pressure Operation, And Transparent (In-Service Inspection)

Molten Fluoride Salts Used in Molten Salt Reactors (Fuel in Coolant; AHTR uses clean salt and solid fuel)





Molten Fluoride Salts Have Been Used for a Century to Make Aluminum in Graphite Baths at 1000°C

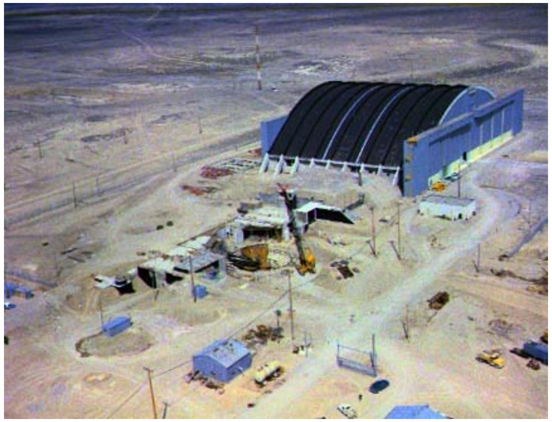
The Billion-Dollar (1950s) Aircraft Nuclear Reactor Propulsion Program Developed Molten Salt Systems for Reactor Applications

[Molten Salt Reactor (Fuel Dissolved In Coolant) For Jet Bomber]

INEEL Shielded Aircraft Hangar

←Hot Cell

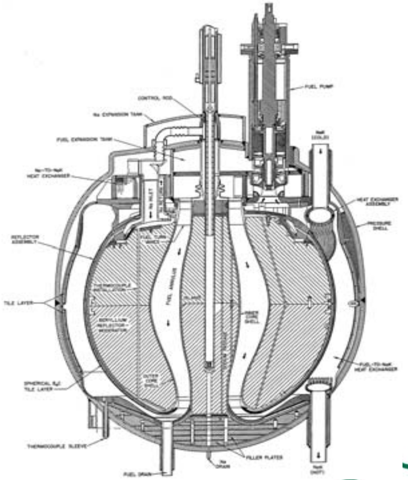
External Views→



ORNL Nuclear Reactor

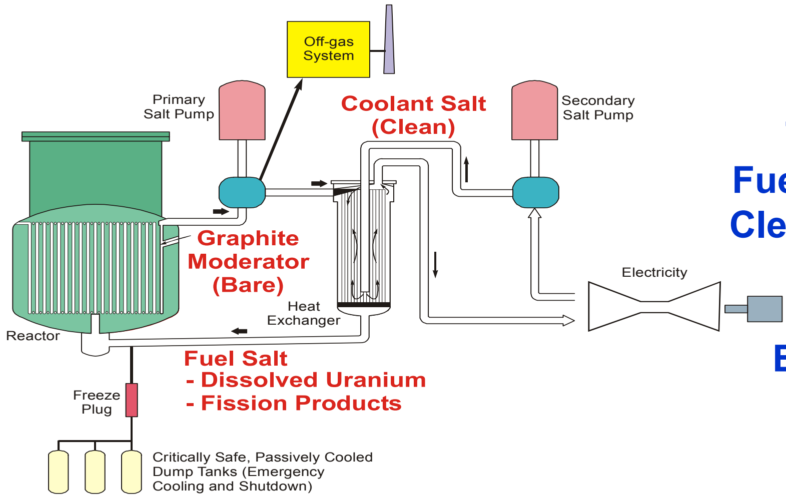
Aircraft Reactor Experiment: First Molten Salt Reactor; 2.5 MW; 882ºC Fuel Salt: Na/Zr Fluoride

Goal: 60 MW(t); 873ºC; Diameter: 56 inch→



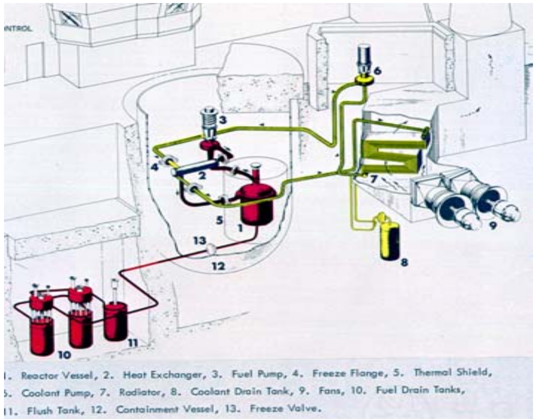
The Molten Salt Breeder Reactor Program Further Developed Molten Salt Technology

(Included Clean Molten Salt Intermediate Heat Transfer Loop)



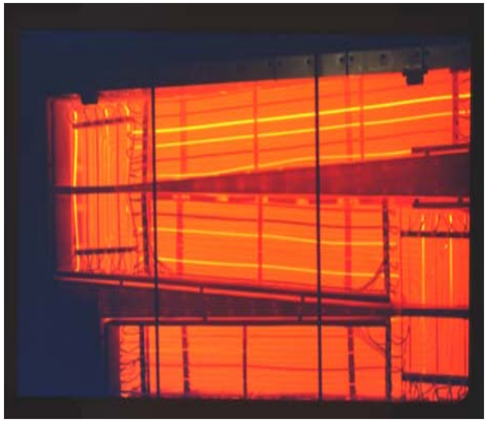
←Concept
T_{exit} = 705ºC
Fuel Salt: ^{7}Li/Be/F
Clean Salt: Na/B/F

Reactor Experimental Building→

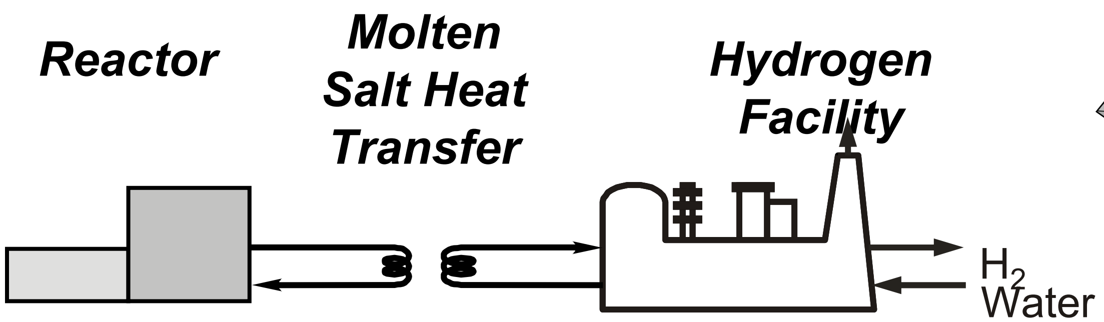


←Test Reactor Layout
Operated at 8 MW(t)
18,000 Hours Critical

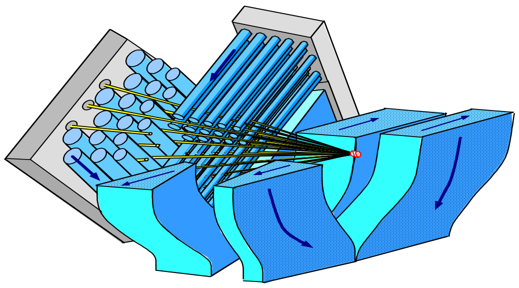
Air-Cooled Heat Exchangers during Operation→



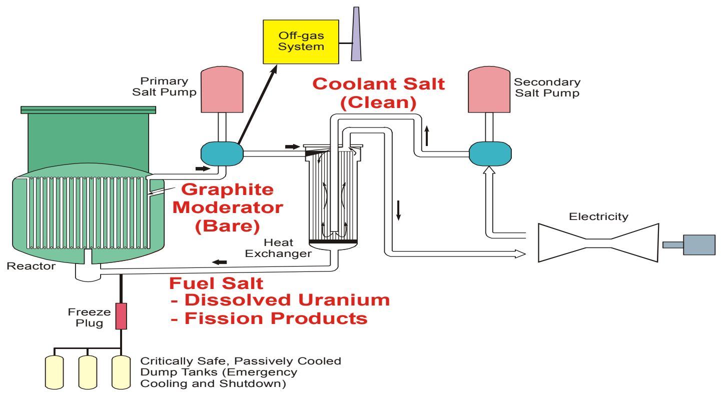
Molten-Salt Technology Is Being Developed for Multiple Future Applications



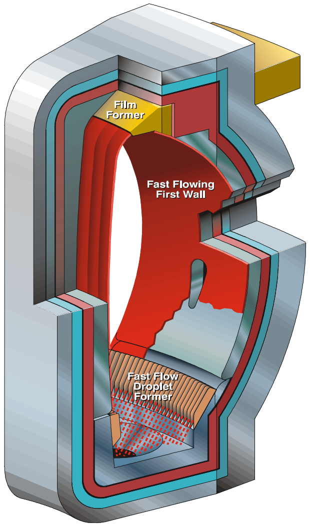
Reactor to Hydrogen Production Facility



Heavy-Ion Inertial Fusion



Molten Salt Reactor (EC and France)



Magnet Fusion Tokamak

Different Reactor Applications Require Different Mixtures of Molten Fluoride Salts

[Aircraft (Na:Zr), Breeder (Fuel: ^{7}Li:Be; Clean Secondary Salt: Na:B)]

Economics
- Capital
- Fuel Cycle
- Maintenance

Corrosion
- Noble Relative to Materials of Construction

Candidate Fluorides for Mixtures
^{7}LiF, NaF, RbF, BeF_2, ZrF_4, AlF_3

Neutronics
- Low Cross Sections
- Activation Dose
  - Gamma
  - Tritium

Physical Properties
- Freeze Point
- Heat Capacity
- Viscosity

Physical Properties of Demonstrated Coolants

Coolant	T_{melt} (ºC)	T_{boil} (ºC)	ρ (kg/m^3)	ρC_p ($kJ/m^{3}{}^{o}C$)	k ($W/m^{o}C$)	v·106 (m^2/s)
$^{7}Li_2BeF_4$ (MSRE)	459*	1,430	1,940	4,540	1.0	2.9
0.58 NaF-0.42 ZrF_4 (ARE)	500*	1,290	3,140	3,670	~1	0.53
Sodium	97.8	883	790	1,000	62	0.25
Lead	328	1,750	10,540	1,700	16	0.13
Water	0	100	732	4,040	0.56	0.13

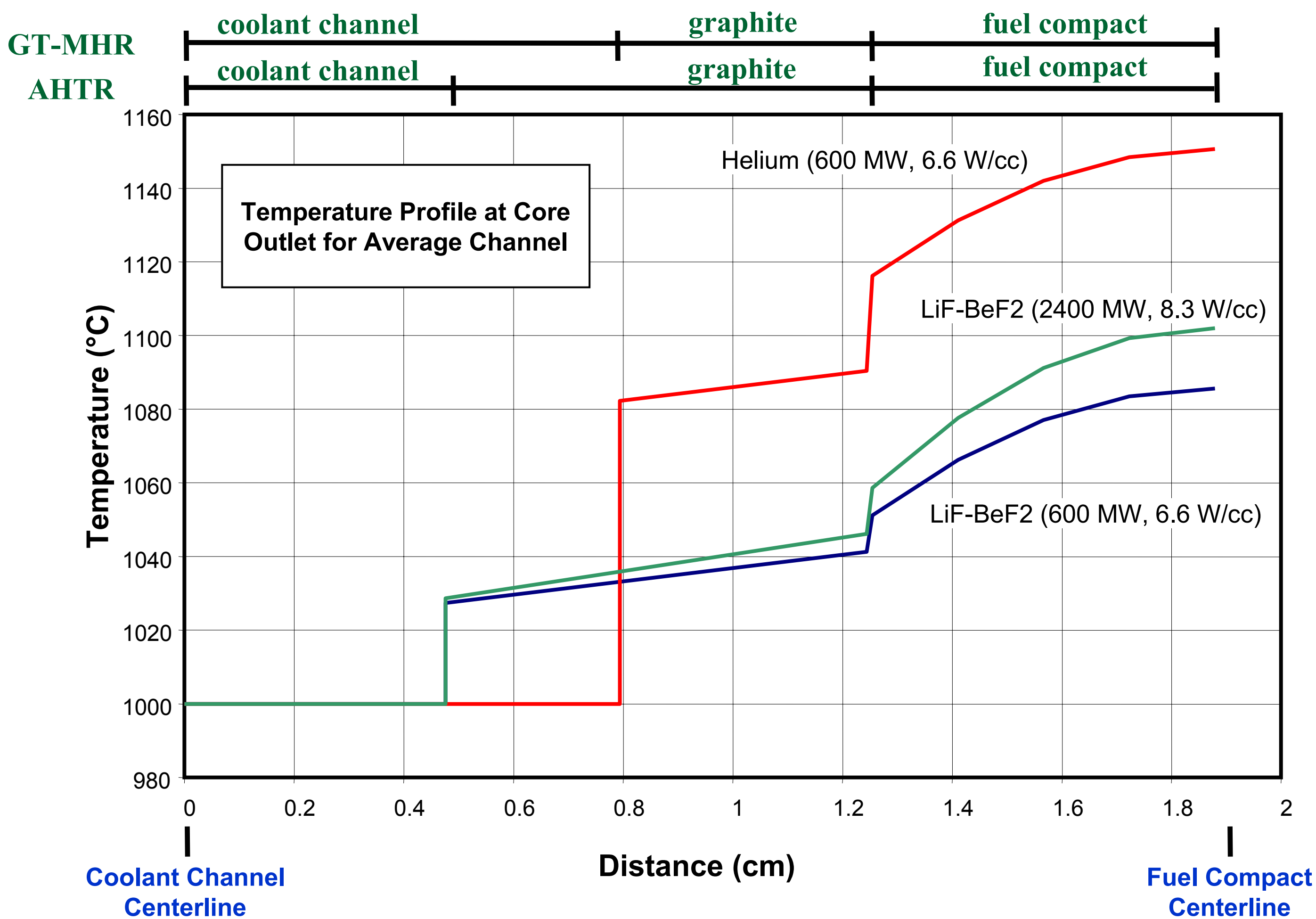
*Salts Used in Reactors. Examples of fluoride salts with lower melting points: Li-Na-Be (22-44-33): ~300ºC; Na-Rb-Zr (6-46-48): 380ºC

Properties at ~700ºC except pressurized water at 290ºC

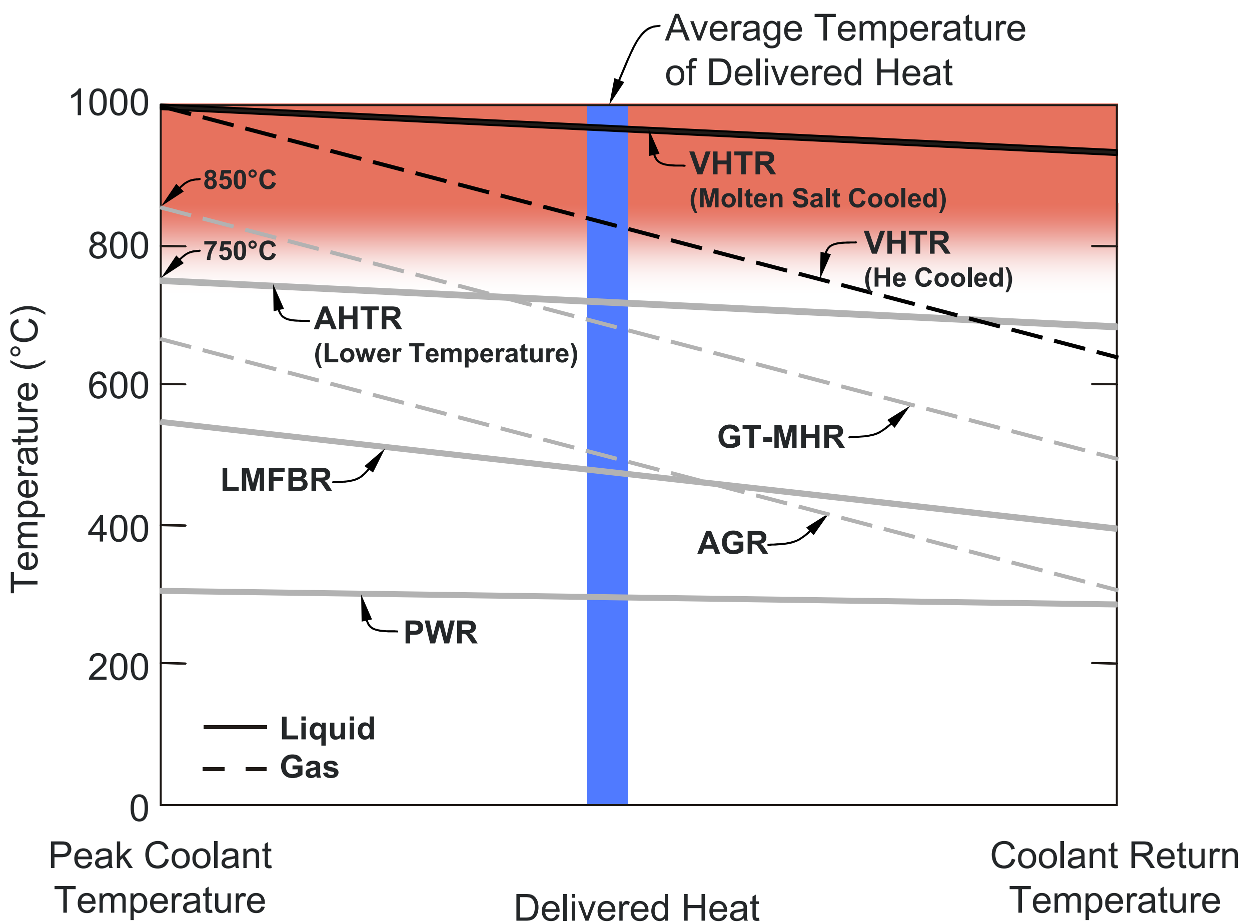
Molten Salts Have Superior Capabilities To Transport Heat At Reactor Conditions

	Water	Sodium	Helium	Molten Salt
Pressure, MPa	15.5	0.69	7.07	0.69
Outlet Temp, ºC	320	545	1000	1000
Velocity, m/s (ft/s)	6 (20)	6 (20)	75 (250)	6 (20)
Number of 1 m dia. pipes required to transport 1000 MW(t) with 100ºC rise at reactor outlet coolant conditions	0.6	2.0	12.3	0.5

Liquids Remove Heat More Effectively Than Gas:
Cooler Fuel For the Same Coolant Exit Temperatures



For Any Coolant Exit Temperature, the Average Temperature of Delivered Heat (the Product) Is Higher with Liquid Coolants than with Gas Coolants



Materials

Graphite and Graphite Fuels

Compatibility with Molten Salt Demonstrated

Metals

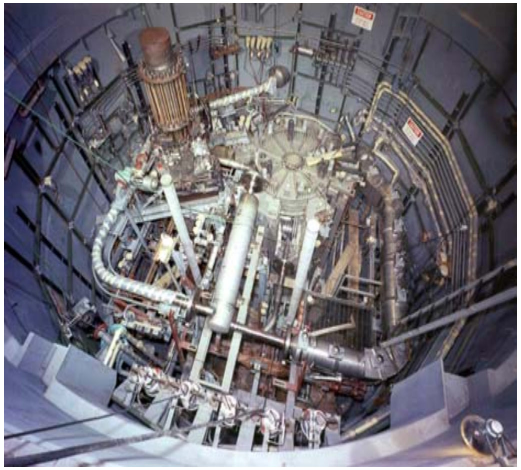
Existing Code Materials to 750ºC

Candidates For Higher Temperature Operation

Status: Similar to Helium-Cooled VHTR, Need to Qualify and Demonstrate Higher-Temperature Materials

Experience Shows that Fluoride Salts Are Compatible with Carbon-Based Materials

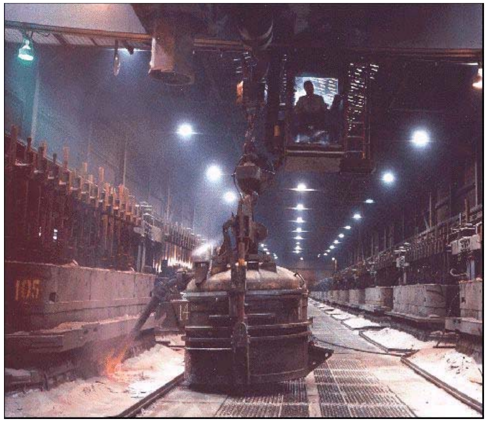
(Same Graphite Radiation Damage Issues As With Gas-Cooled Reactors)



Molten Salt Reactor Experiment (8 MW(t))

←Reactor Compartment

Graphite Core (Moderator)→



←Aluminum Plant: 1000ºC (NaF-ALF_3 Molten Salt in Graphite)

Molten Salt Reactor Experiment [8 MW(t)] Postirradiation Graphite→

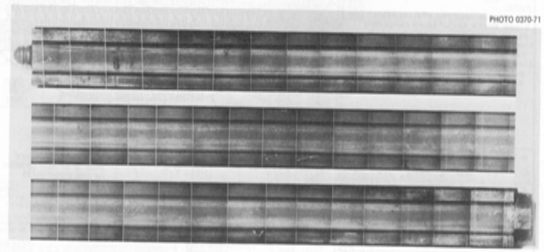


Fig. 6. One of the removable graphite moderator elements after operation in the MSRE, showing the fuel channel facing the center of the reactor. Machining marks are plainly visible in the fuel channel, showing excellent condition of the core block.

Good Candidates Exist for High-Temperature Metallic Components for AHTR

- Piping, valves, heat exchangers must function for long times at 1000°C in contact with both air and molten salt
- Pumps will need to survive to higher temperatures for short times and resist molten salt corrosion
- Materials considerations for service in AHTR
  - Corrosion, mass transfer, strength (long-term and short-term), thermal aging & embrittlement, irradiation degradation, fabricability, experience base & maturity, codification
- Stable high-strength, high-temperature materials with salt-resistant nickel coatings will likely work
  - Inconel 617, Haynes 230, Alloy 800 H, Hastelloy X or XR, VDM 602CA, HP modified, etc.
- Monolithic materials will require high nickel for salt resistance, plus sufficient high-temperature strength
  - Haynes 214, Cast Ni-based superalloys, ODS alloys (MA 754 and 956)
  - Alloys typically much less mature, hence codification will be require

Good Candidates Exist for High-Temperature Reactor Vessel Materials for the AHTR

- **Pressure vessel must function for long times at 500°C**
- **Must survive higher temperatures (≈800°C) for short times (≈100hr) during accidents**
- **Considerations for service in AHTR**
  - Corrosion, mass transfer, strength (long-term and short-term), thermal aging & embrittlement, irradiation degradation, high emissivity, fabricability, experience base & maturity, codification
- **Stable, high-strength, high-temperature materials with salt-resistant nickel coatings will likely work**
  - Advanced ferritic-martensitic steels have sufficient strength for normal operation–9Cr-1MoV
  - Excessive off-normal temperatures may require higher temperature alloys–304L, 316L, 347, Alloy 800H or HT
- **Monolithic materials require high nickel for salt resistance, plus sufficient high-temperature strength**
  - Alloy 800H or HT, Hastelloy N, Haynes 242
  - Codification may be required for specific product forms

Facility Design

AHTR: A Low-Pressure,

High-Temperature Liquid-Cooled Reactor

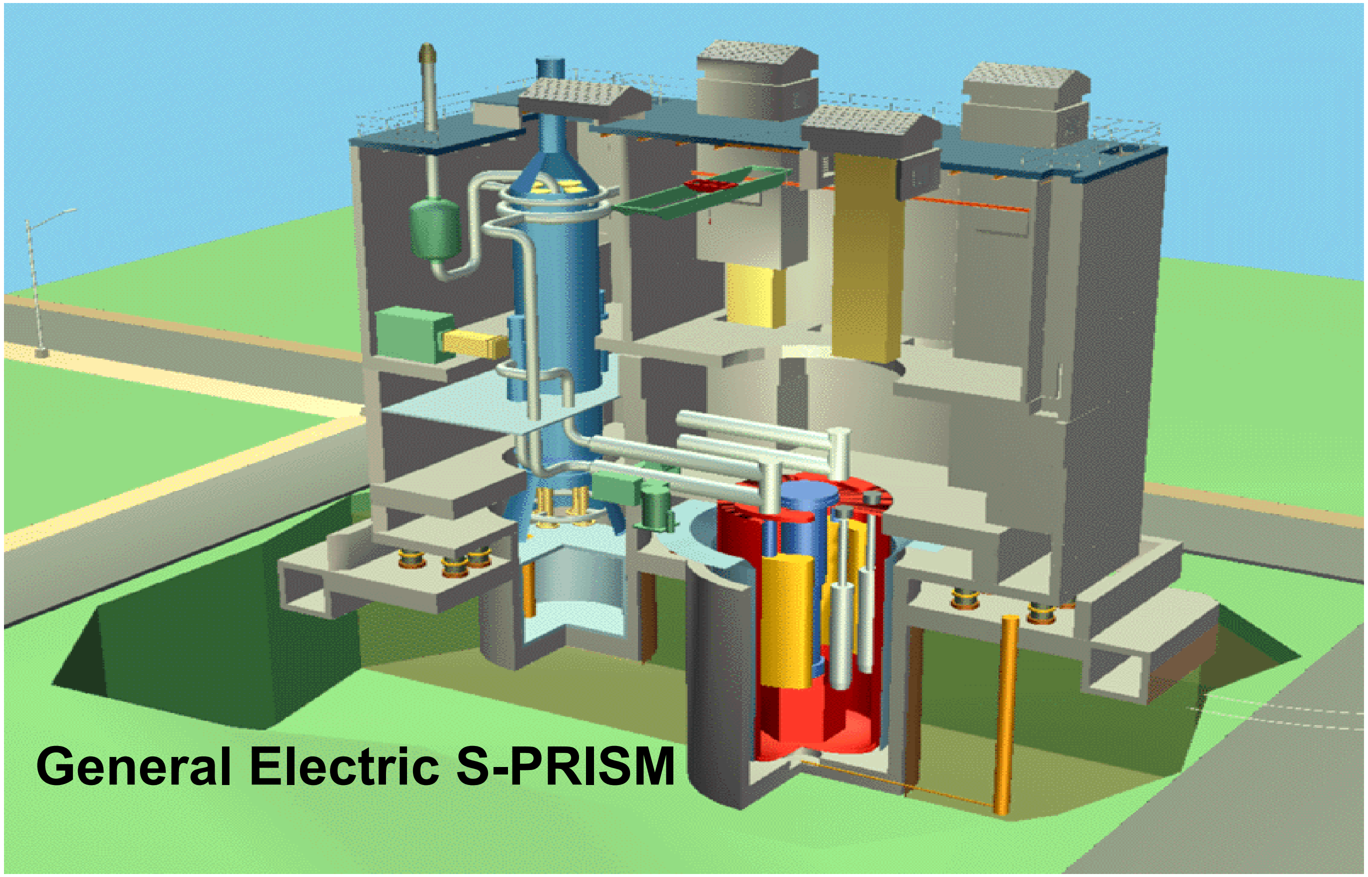
Not a Point Design

Current AHTR Parameter Set

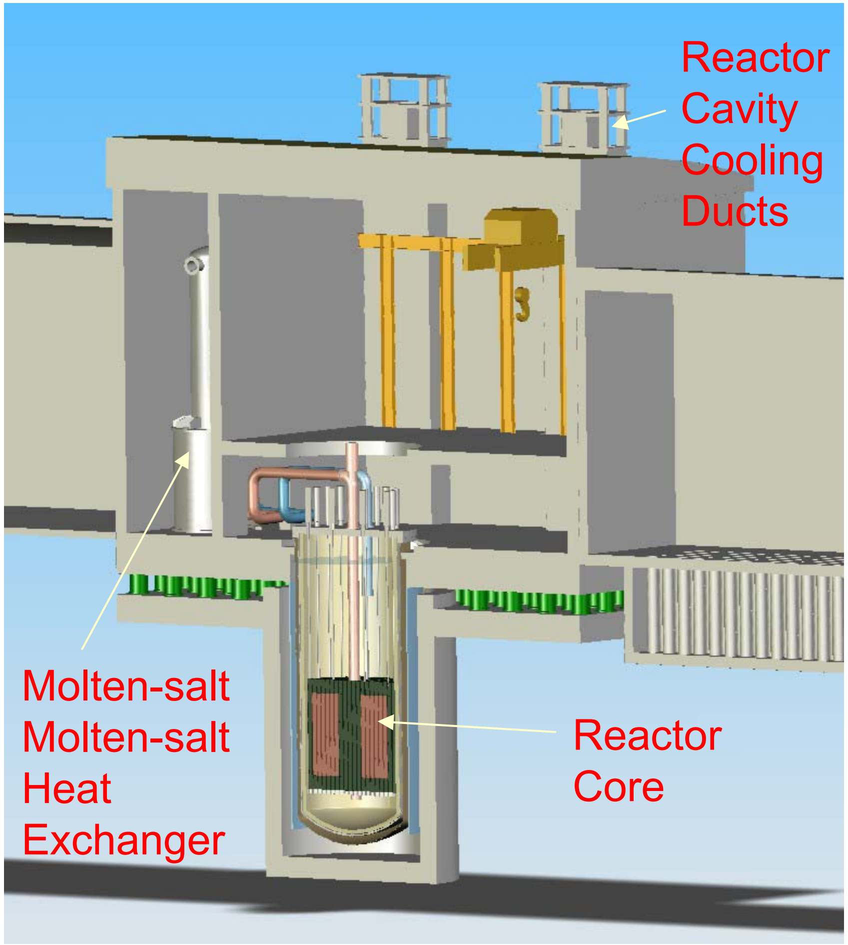
Power lever	2400 MW(t)		Core diameter	7.8 m
Core inlet/outlet temperature	900ºC/1000ºC		Core height	7.9 m
Core inlet/outlet pressures	0.230MPa/0.101 MPa		Fuel annulus	2.3 m
Core pressure drop	0.129 MPa (18.7 PSI)		Number of fuel columns	324
Coolant (several options)	2LiF-BeF$_2$		Number inner reflector columns	55
Coolant mass flow rate	12,070 kg/s (20% bypass)		Number outer reflector columns	138
Coolant volumetric flow rate	5.54 m^3/s		Fuel kernel	UCO
Coolant channel diameter	0.95 cm		Fuel enrichment	10.36 wt% ^{235}U
Coolant fraction (active core)	6.57%		Mean core power density	8.3 W/cm^3
Coolant velocity	2.32 m/s (7.6 ft/s)		Peak core power density	TBD W/cm^3
Core pumping power	716 kW		Reactivity, temperature	Negative Doppler
Vessel Pressure	~0.1 MPa		Reactivity, void (whole core)	TBD
Vessel external temperature			Mean fuel temperature	1050ºC
Vessel outside diameter	9.2 m		Peak fuel operating temperature	1168ºC
Vessel wall thickness	5-10 cm (material dependent)			
Net electric output	~1200 MW(e)			
Net plant efficiency	48-59%			

Proposed AHTR Facility Layouts Are Similar to Sodium-Cooled Fast Reactors

Low-Pressure, High-Temperature, Liquid-Cooled

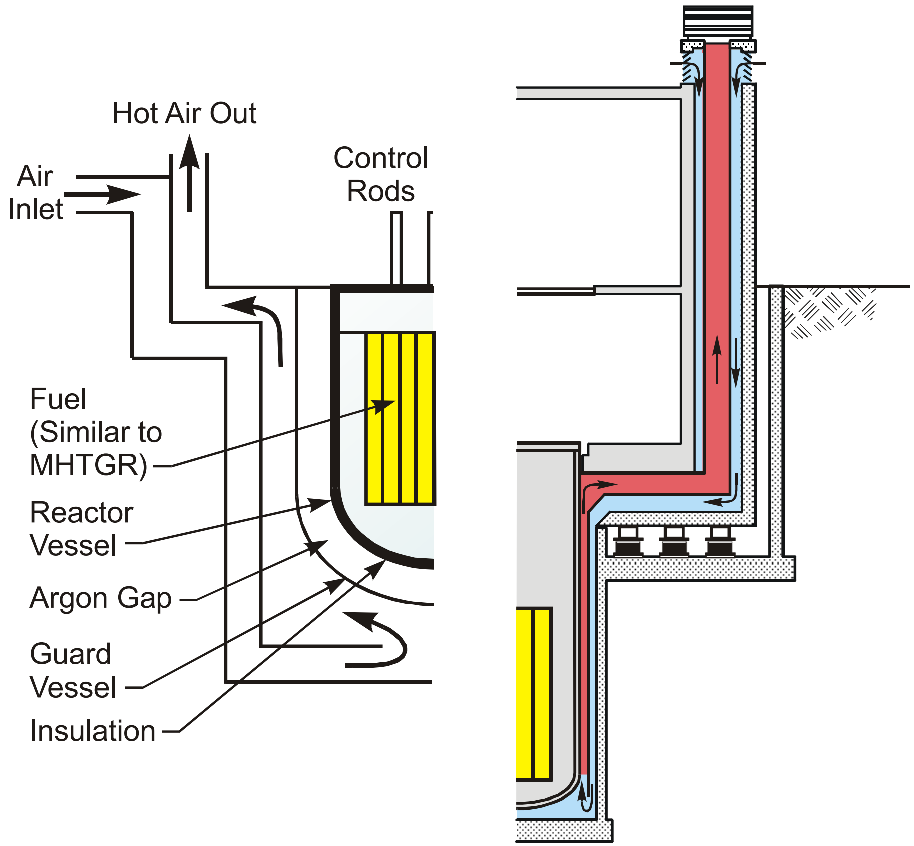


AHTR [2400 MW(t)] Nuclear Island Is Similar Size To S-PRISM [1000 MW(t)]



- **Differences from S-PRISM facility layout:**
  - **No SNF storage in vessel**
  - **No heat exchanger inside vessel**
  - **Molten salt-to-gas heat exchanger in turbine hall**
- **Same vessel size**
  - **Space for 2400 MW(t) AHTR core with low power density**
- **Similar Equipment Size**
  - **Molten salt volumetric heat capacity > sodium**
- **Higher capacity decay heat removal system**
  - **Higher vessel temperatures**
- **Higher electrical output**
  - **S-PRISM: 380 MW(e)**
  - **AHTR: >1200 MW(e) Higher temperatures)**

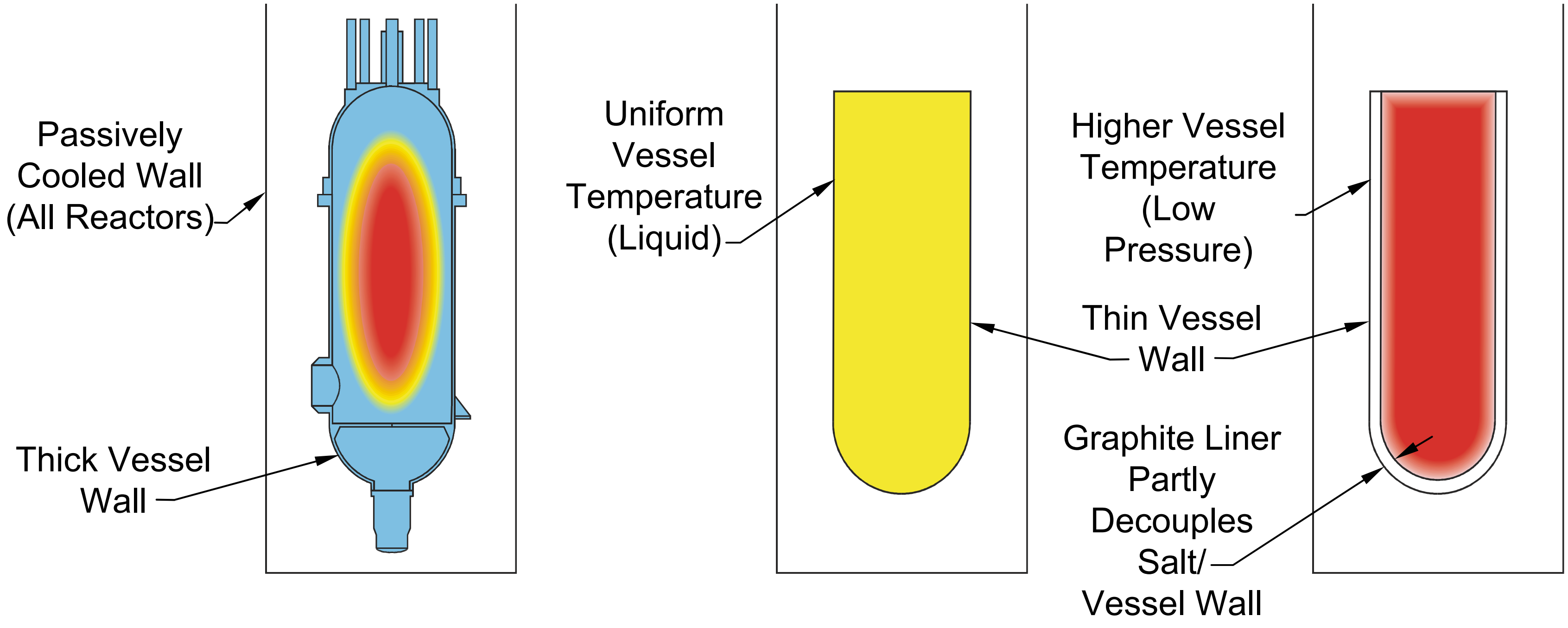
In an Emergency, Decay Heat Is Transferred to the Reactor Vessel and Then to the Environment



- Similar to GE S-PRISM (LMR)
- Argon Gap
  - Heat Transfer ~T^4
  - Thermal Switch Mechanism
- Heat Rejection: Temperature Dependent
  - LMR: 500-550ºC [~1000 Mw(t)]
  - AHTR: 750-1000ºC [>2000 Mw(t)]
- High Heat Capacity
  - Molten Salt and Graphite
  - High Temperature (Limited-Insulation of Vessel from Hot Salt)

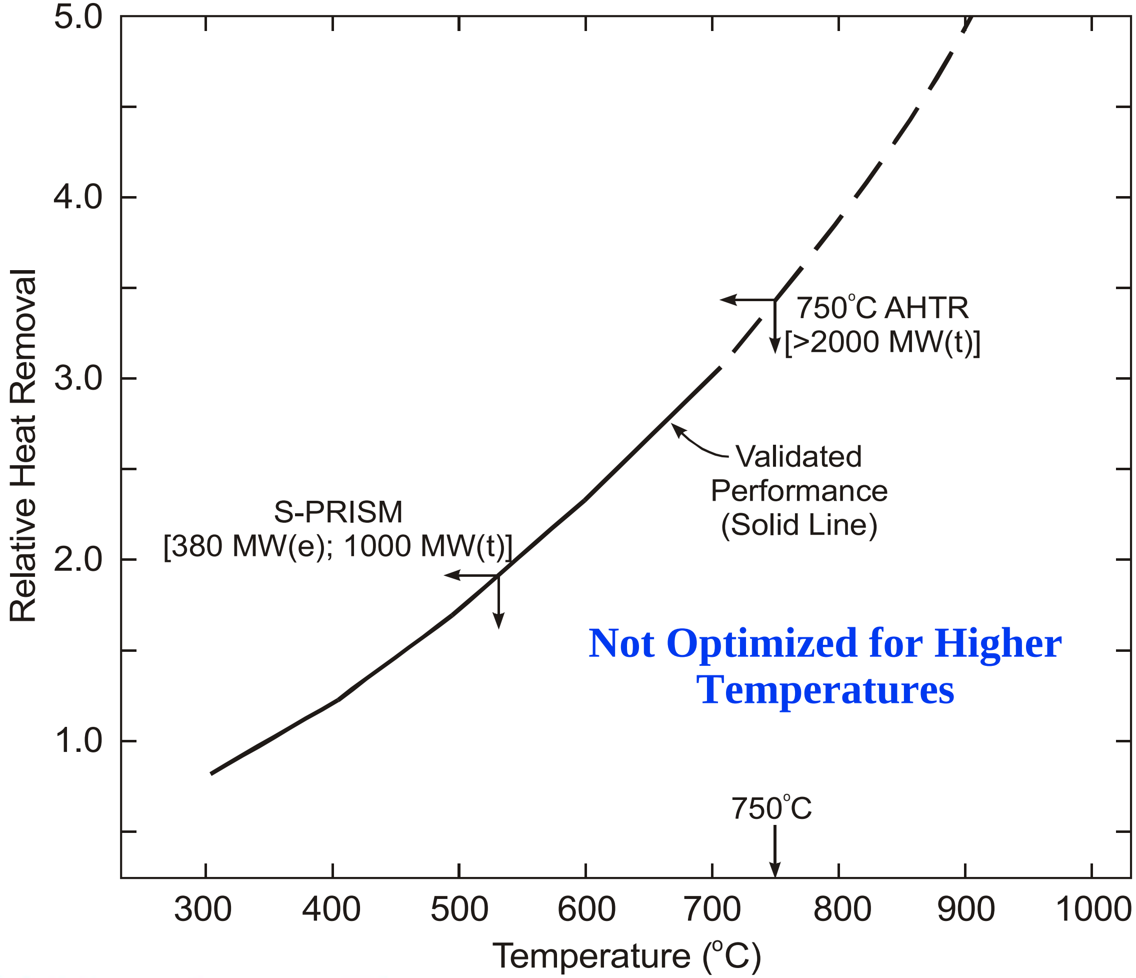
High-Temperature Low-Pressure Liquid Coolants Enable the Design of Large Reactors with Passive Safety

Reactor	MHTGR	S-Prism	AHTR
Coolant	Helium	Sodium	Molten Salt
Power	600 MW(t)	1000 MW(t)	>2000 MW(t)
Limitation	Conduction	Sodium Boiling Point	Vessel



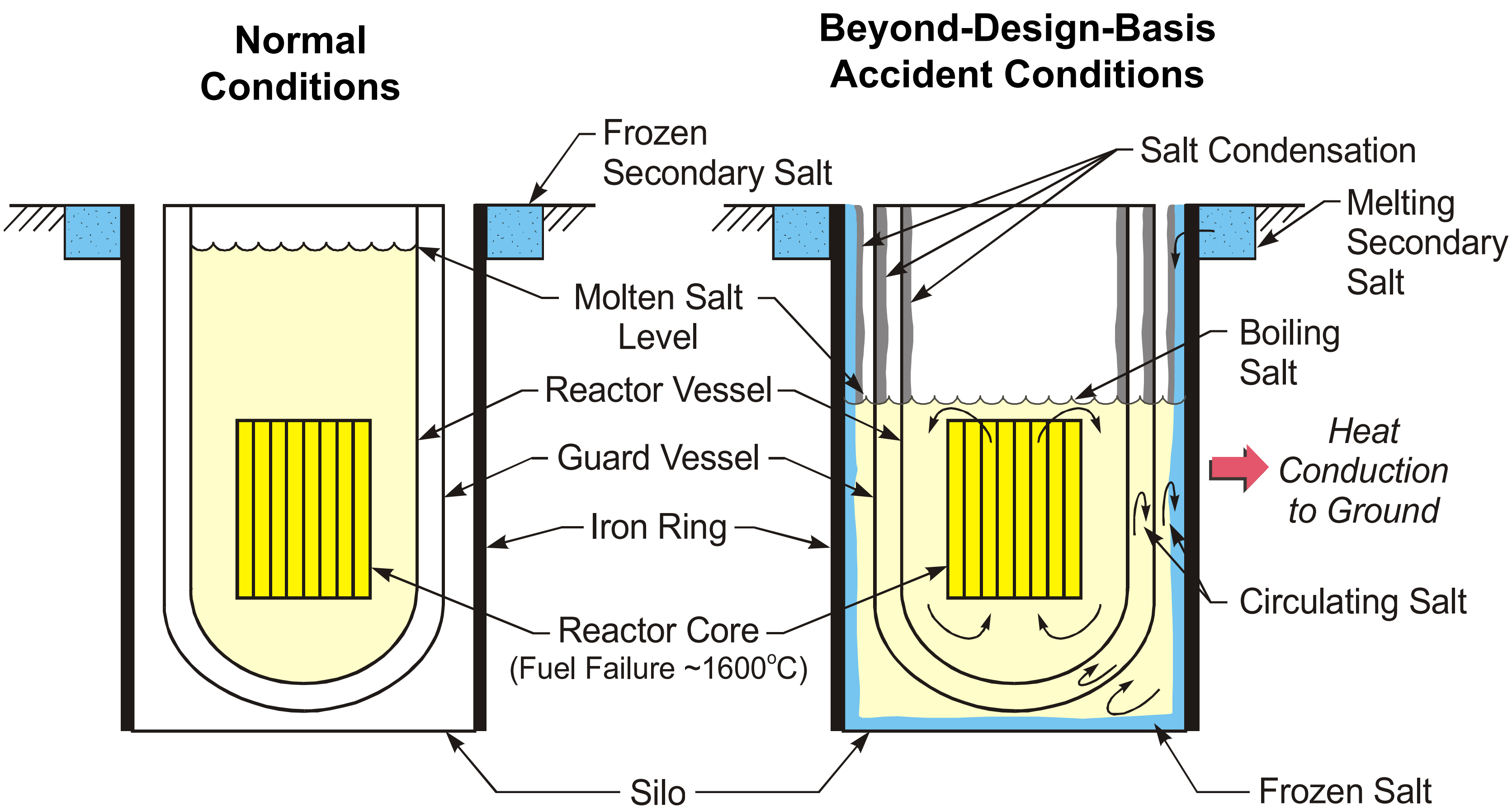
Decay Heat Removal Increases Rapidly with Temperature

(S-PRISM Vessel with Air-Cooled System)



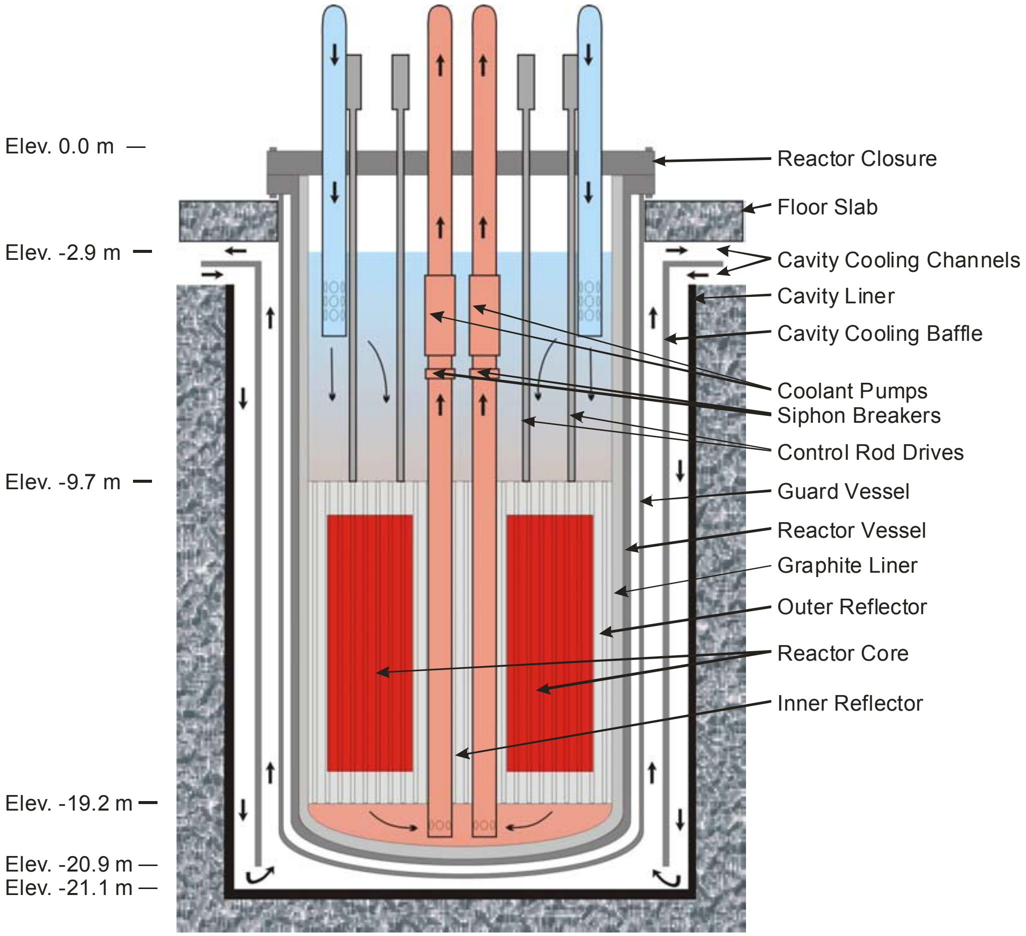
Beyond-Design-Basis Accident Avoids Radionuclide Release By Multiple Mechanisms

Molten Salts Trap Radionuclides (Including Cs and I) in the Salt, Isolate SNF from Air, Can Not pressurize Containment, and Transfer Heat to the Silo If Vessel Failure

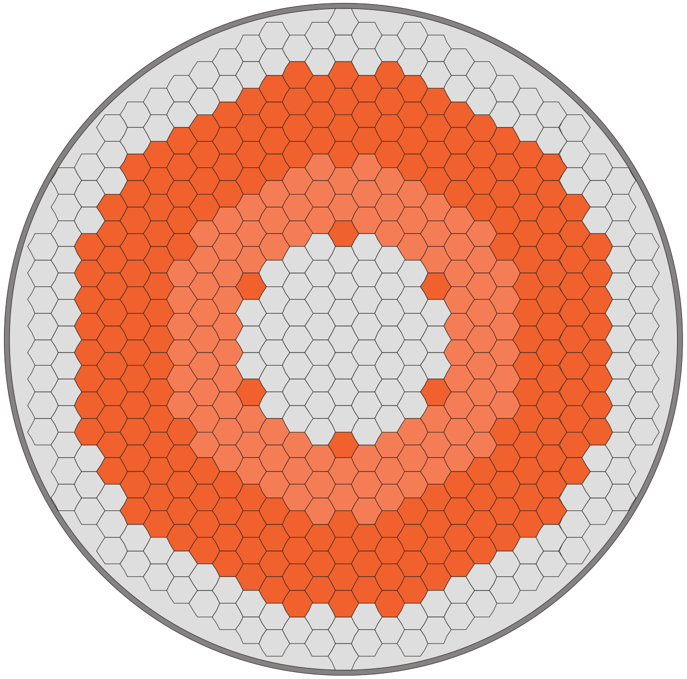


Reactor Core Design

AHTR 9.0m Vessel Allows 2400 MW(t) Core

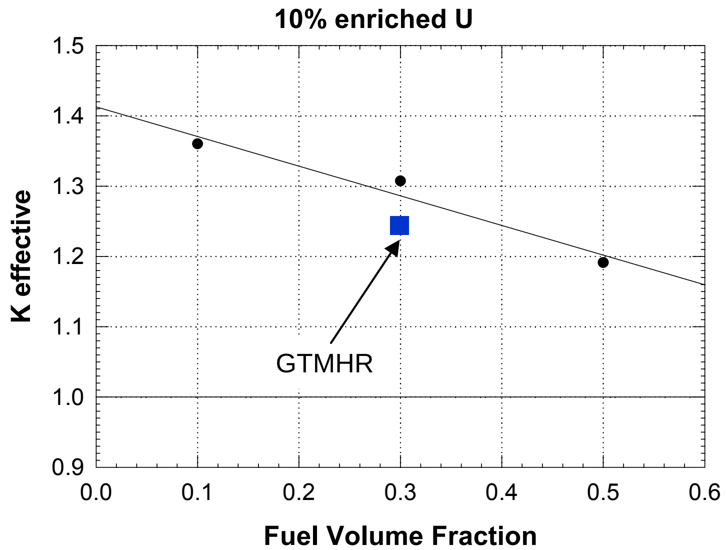


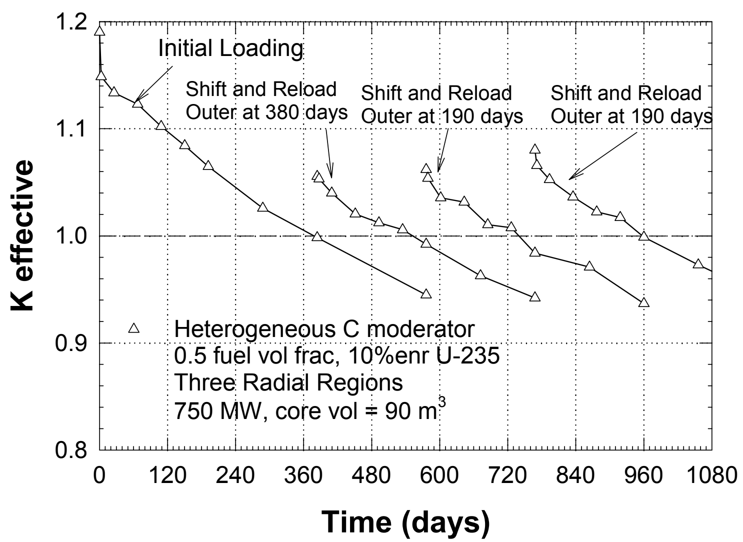
102 Original fuel columns
<u>222 Additional fuel columns</u>
324 Total fuel columns



Power density = 8.3 MW/m^3
(26% larger than 600 MW GT-MHR)

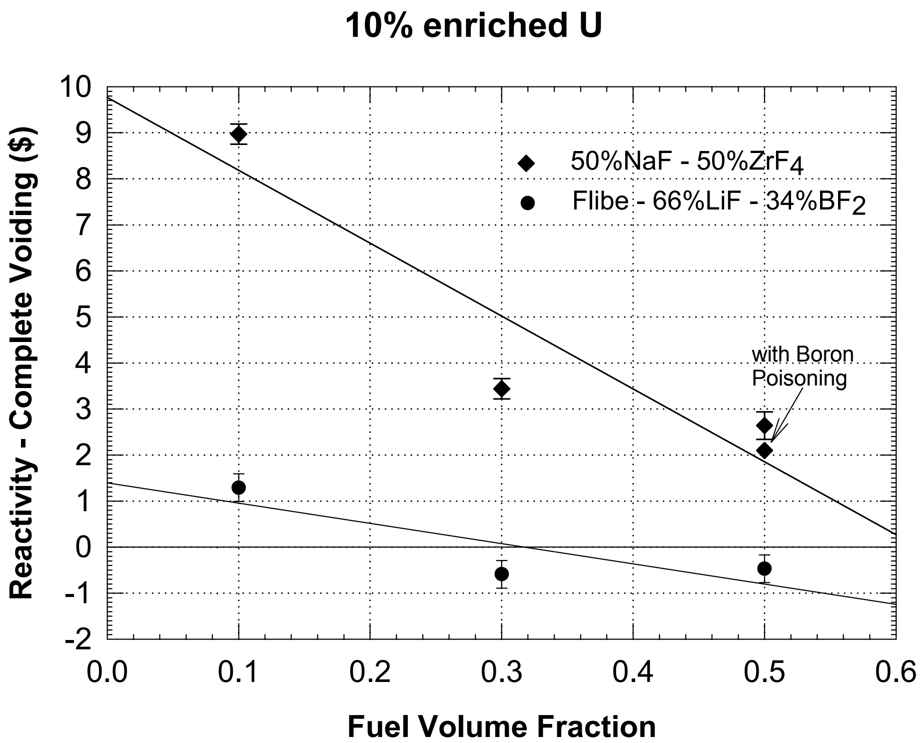
AHTR And Gas-Cooled Reactors Have Similar Neutronics





- Excess reactivity similar for given core loading
- Lower coolant volume fraction
- Neutron lifetime ~1ms
- k_{eff} increases with higher moderator to fuel ratio (undermoderated in design region)
- Large negative temperature feedback due to Doppler effects (~ -$0.01/K)
- Similar fuel burnup/ fuel cycle options
- At 8.3 W/cm^3, core life is ~ 580 days
  - 10% enrichment
  - 0.5 fuel volume fraction

AHTR Void Coefficient Depends on Salt Composition and Configuration

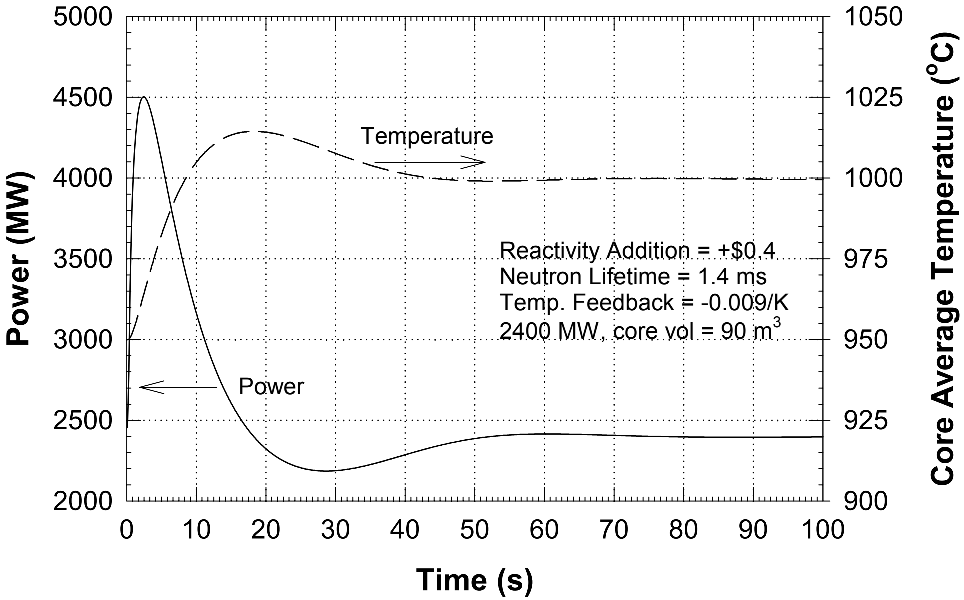


SNL Model

- Void coefficient ranges from negative to positive, depending on coolant fraction and salt choice
- 66%^{7}LiF/33%BeF_2 negative for higher fuel fractions
- 50%NaF/50%ZrF_4 most positive of salt options
- 1st order effect is absorption, 2nd order effect is moderation
- Ranking – Be,^{7}Li, Mg, Zr, Na
- Design options may allow use of several salts
  - Coolant fraction
  - Heterogeneous core design
  - Burnable absorbers
  - Isotopic purity
  - Fuel loading/enrichment

Implications of AHTR Designs With Positive Void Coefficients

Example: Na-Zr Salt (worst salt) with 20% Flow Blockage: +$0.40 Instantaneous Reactivity Insertion



- Core power increases but is mitigated by increase in fuel temp of ~60ºC.
- Relatively slow transient (10's of sec).
- Core reaches lower equilibrium power – issue is heat-up of blocked fuel columns (~9 ºC/sec)

- Effects can be mitigated by:
  - Salt Composition
  - Coolant Volume
  - Burnable Poison
  - Fission Produce Poisoning
  - Fuel Fraction
  - Enrichment
  - Core Geometry
    - Parfait
    - Heterogeneous cores

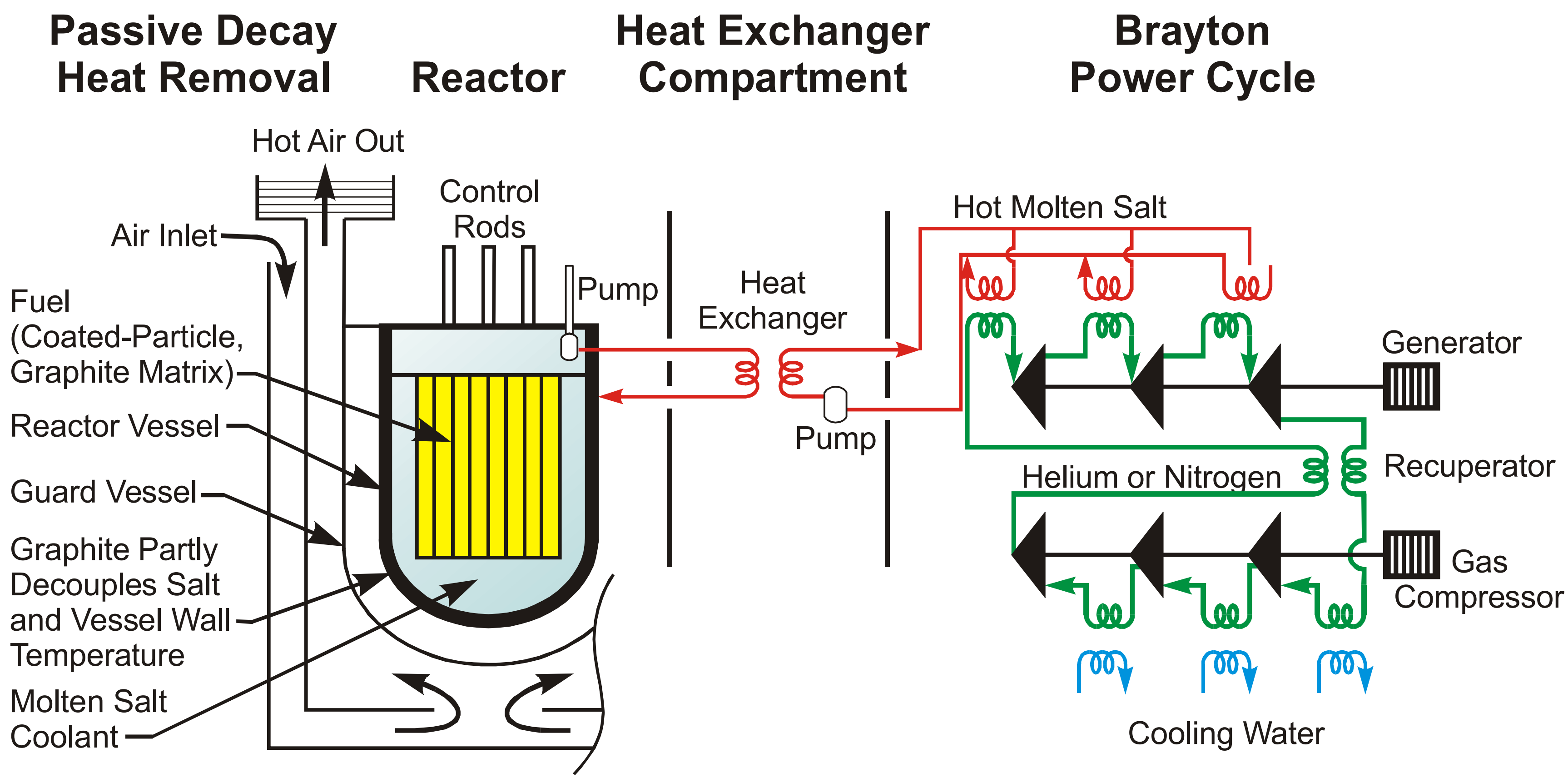
- Positive effects are limited in reactivity, and would be confined to local effects due to coolant channel blockage

Energy Conversion

Electricity

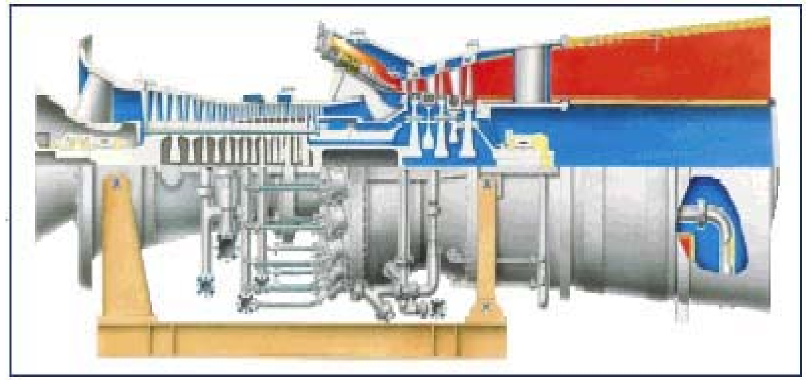
Hydrogen

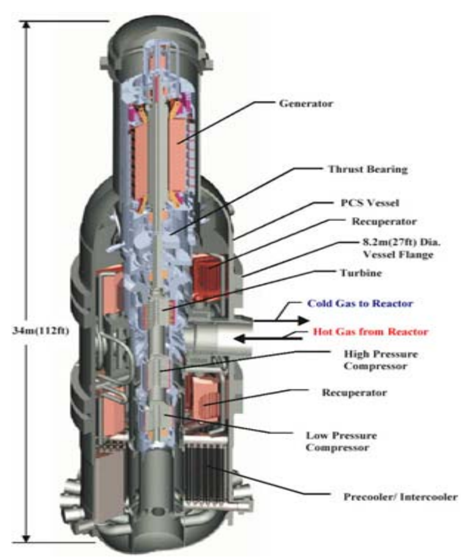
A *Multi-Reheat* Brayton Cycle Is Used for Efficient Electricity Production



Near Term: Use Nitrogen (with Minor Amounts of Helium) Brayton Cycle

- **Same turbine technology as existing natural-gas-fired turbines**
  - **Lower temperatures than current commercial units**
  - **Option for small amounts of helium to improve thermal properties (reduce heat exchanger size)**
- **Helium Brayton Cycle**
  - **Second-generation option**
  - **Use if developed for helium-cooled reactor**
- **Minimize technical risk and development cost with little penalty**

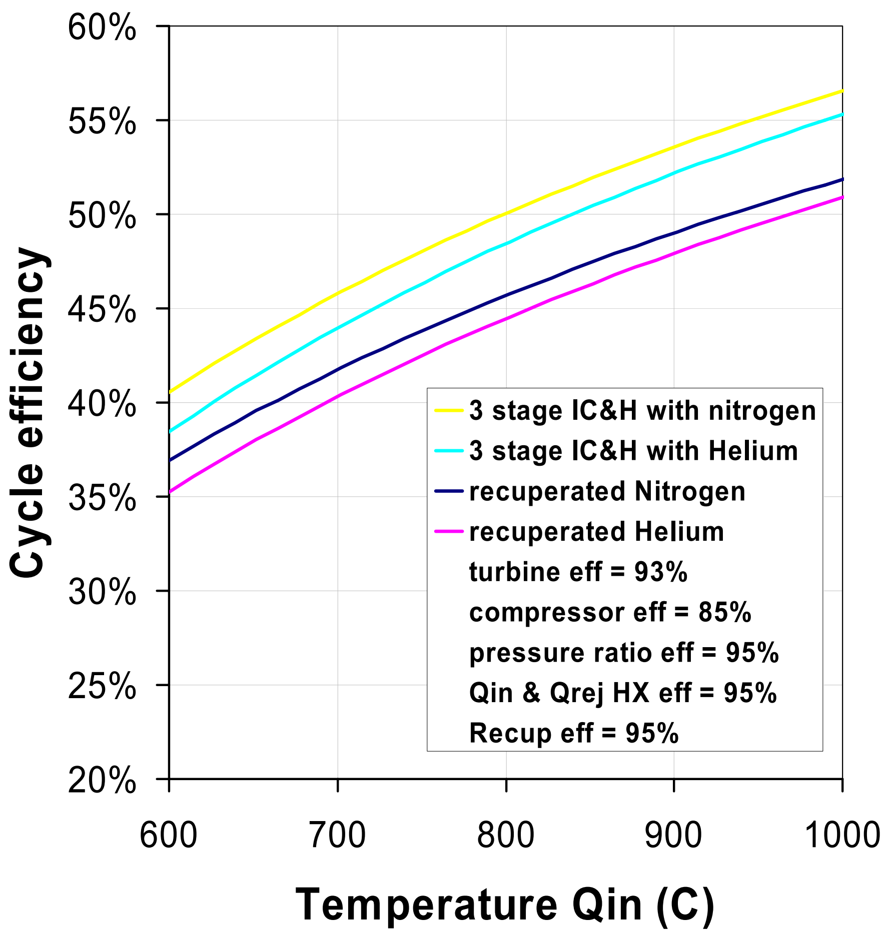


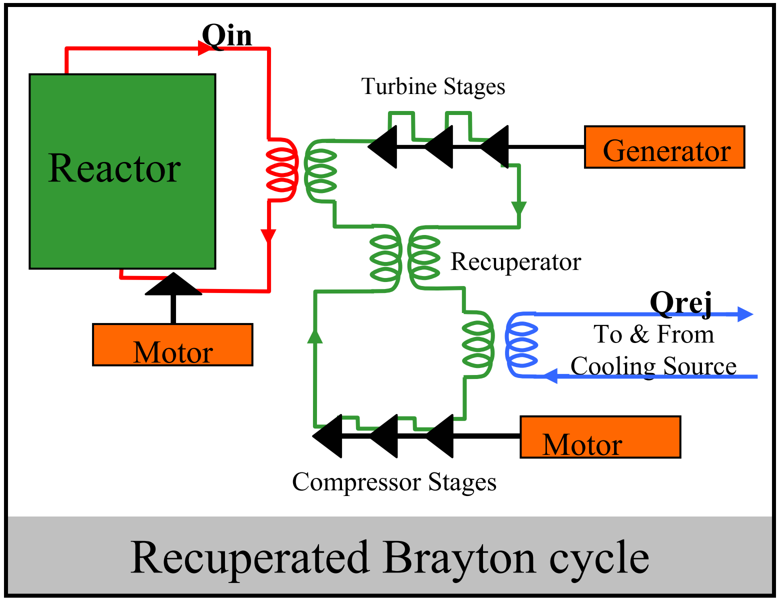


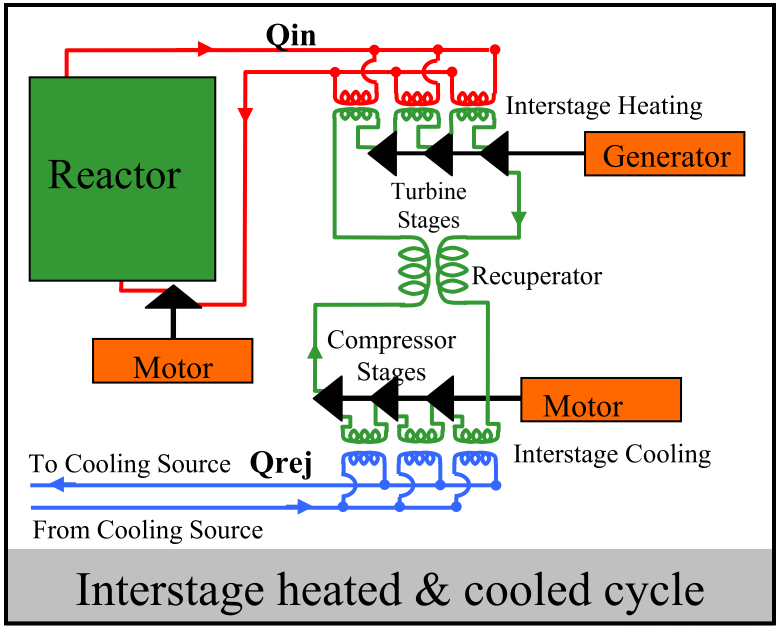
Above: GE Power Systems MS7001FB

Left: GT-MHR PCU (Russian Design)

Nitrogen and Helium Brayton Cycle Options can be Considered for AHTR

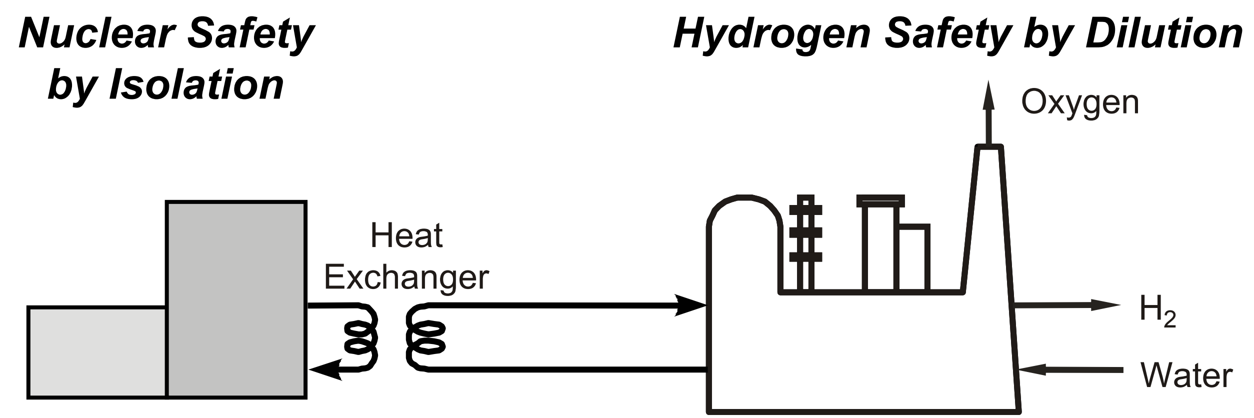






Hydrogen Production May Require Development of Molten Salt Coolant Technology

(AHTR Better Couples to Such Systems)



- **Smaller system (1 molten salt loop = 25 Helium loops)**
  - **Lower heat loses**
  - **Lower costs**
- **Chemical plant safety (German chemical industry evaluation)**
  - **No compressed-gas energy**
  - **Avoid toxic chemical release if heat exchanger failure**

ECONOMICS

Larger Reactors Have the Potential for Lower Capital and Operating Costs

Initial Cost Analysis Shows That AHTR Economics Is Favorable

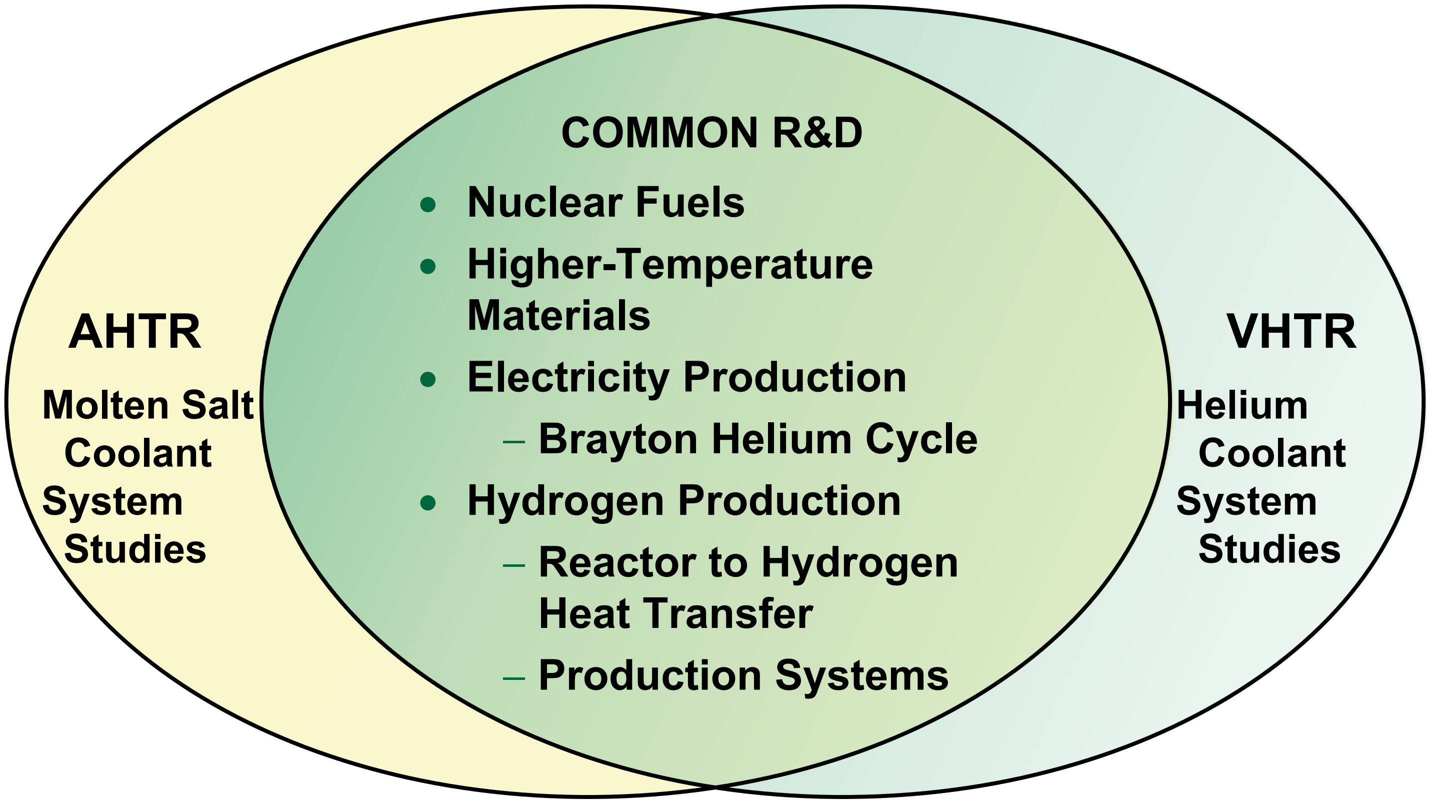
- Used scaled cost data from S-PRISM and GT-MHR
  - S-PRISM for reactor, construction, engineering, and contingency
  - GT-MHR for power conversion and heat rejection systems
- Included several cost elements for reactor, power conversion, and balance of plant systems
- Standard scaling laws used to normalize all three reactors to 2400 MW(t)
- Indicates that AHTR will be 73-75% of S-PRISM and GT-MHR costs (large uncertainties)

Research And Development

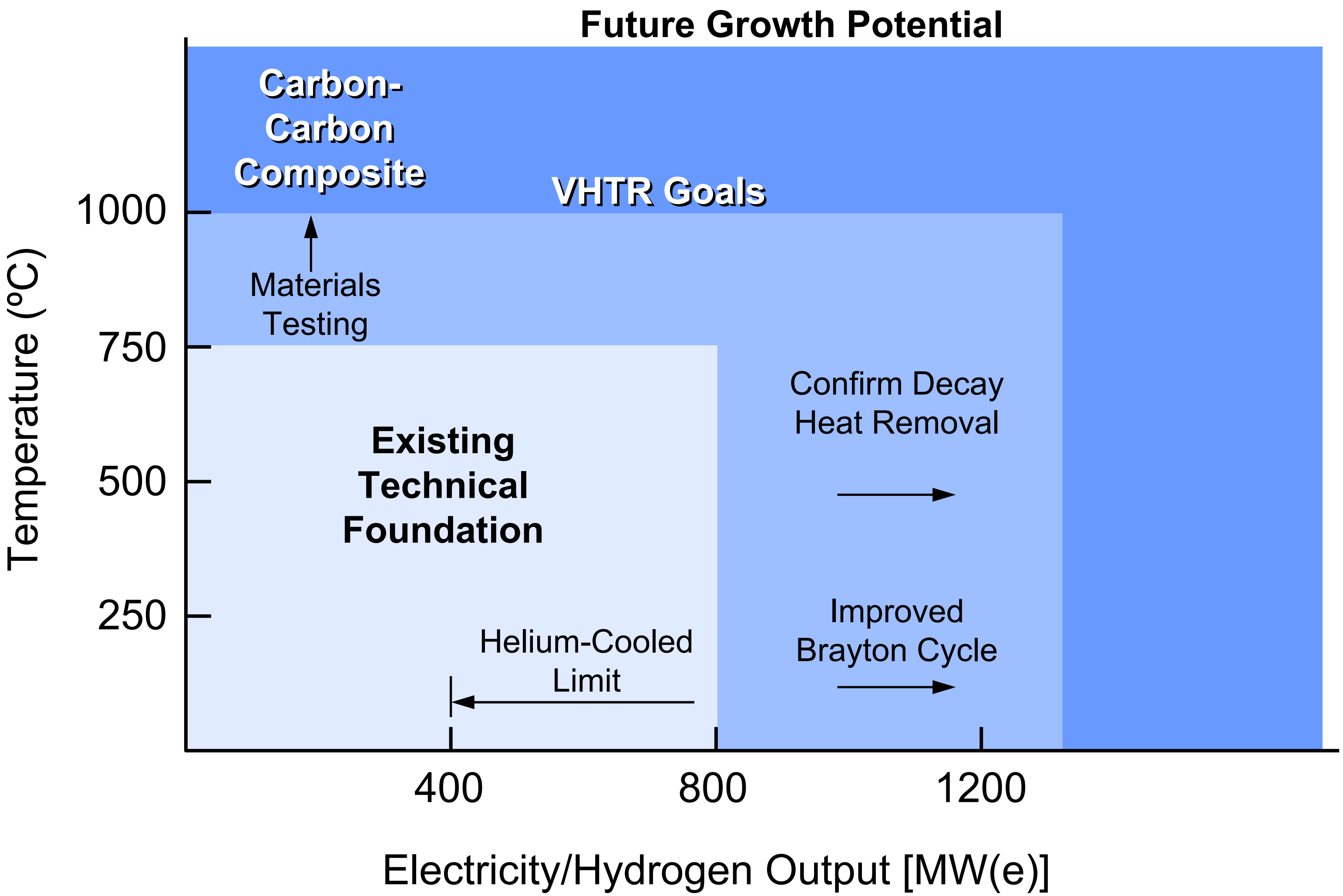
Massive Overlap with Other High-Temperature Reactors

Some Unique Issues

The R&D Requirements for the Molten-Salt-Cooled AHTR and Helium-Cooled VHTR Have Much In Common: Short List of Unique Issues

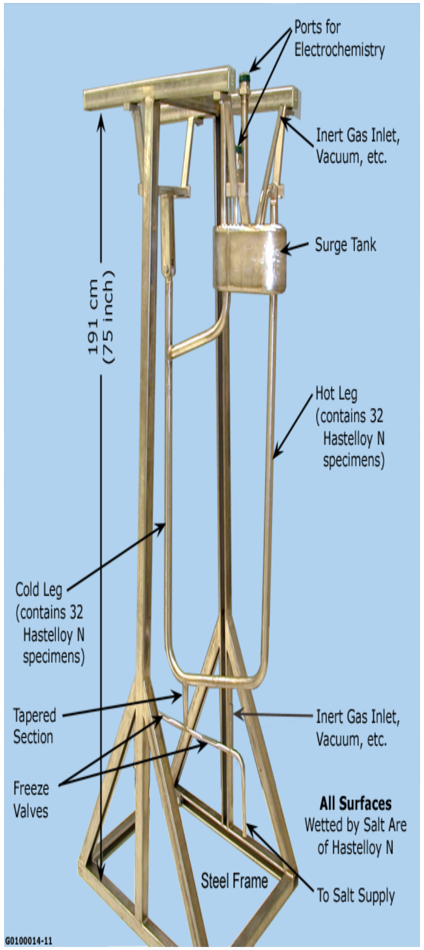


Global AHTR R&D Perspective



Materials Are the Primary Challenge

- **Metals**
  - **Available to 750°C**
  - **Require testing and qualification of materials to >1000°C in molten salt flow loops**
  - **Multi-year effort**
- **Carbon-carbon composites**
  - **Option for heat exchangers?**
- **Fuels**
  - **Same developmental issues as for helium-cooled VHTR**



3000 hours at 815ºC with minimal corrosion by fluoride salt (Williams: Global 2003)

Molten Salt Trade Studies Required To Define The Optimum Composition

Economics
- Capital Cost
- Fuel Cycle
- Maintenance

Corrosion
- Redox Control
- Metal Selection

$^{7}LiF:BeF_2$
$^{7}LiF:ZrF_4$
$^{7}LiF:RbF:ZrF_4$
$NaF:RbF:ZrF_4$
Etc

Neutronics
- Core Analysis
- Activation Products
  - Gamma
  - Tritium

Physical Properties
- Freeze Point
- Heat Capacity
- Viscosity

Several Other Unique R&D Areas

- High-temperature performance of decay-heat cooling systems
  - Controls ultimate size of the reactor
  - Several options
- Systems to avoid salt freezing
  - Chemical industry practice
  - Molten-salt-fueled reactor experience and testing
  - Lead- and sodium-cooled reactor practice
- Plant design (Integration of multiple technologies)
- Choice of demonstration plant size

AHTR Conclusions

- Attractive features compared to gas-cooled reactors
  - Increased power output (economics)
  - Improved compatibility with hydrogen production
  - Increased flexibility in choice of Brayton cycles
  - Reduced fuel temperatures and requirements
  - Improved capture/retention of fission products
- Challenges
  - Point design
    - Salt selection
    - Decay heat removal system
    - Core design
  - Materials for higher-temperature operations
  - High melting temperatures

NOTES/BACKUP

Notes In Same Sequence As Talk

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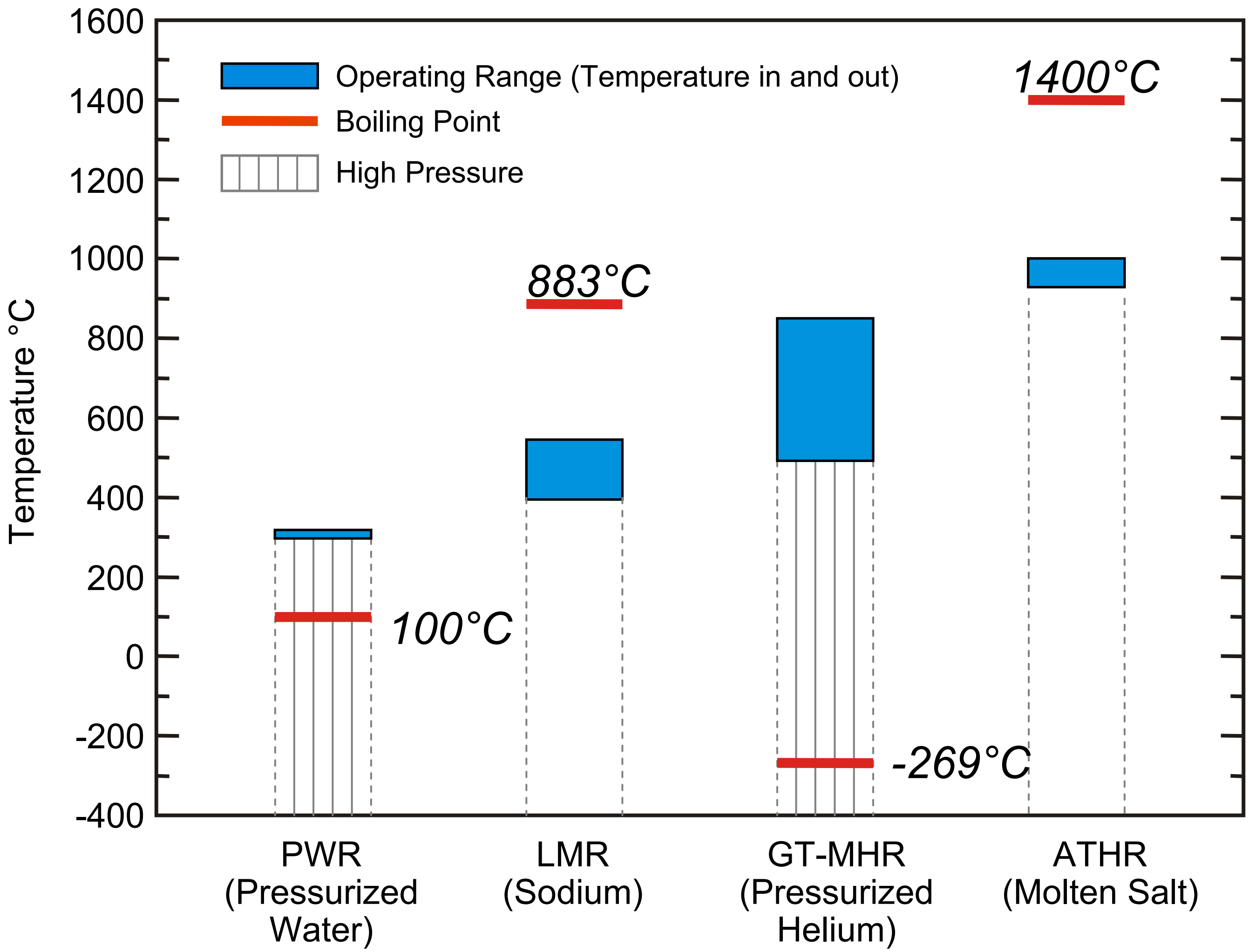
The AHTR Operates at Atmospheric Pressure with Small Temperature Drops across the Reactor Core

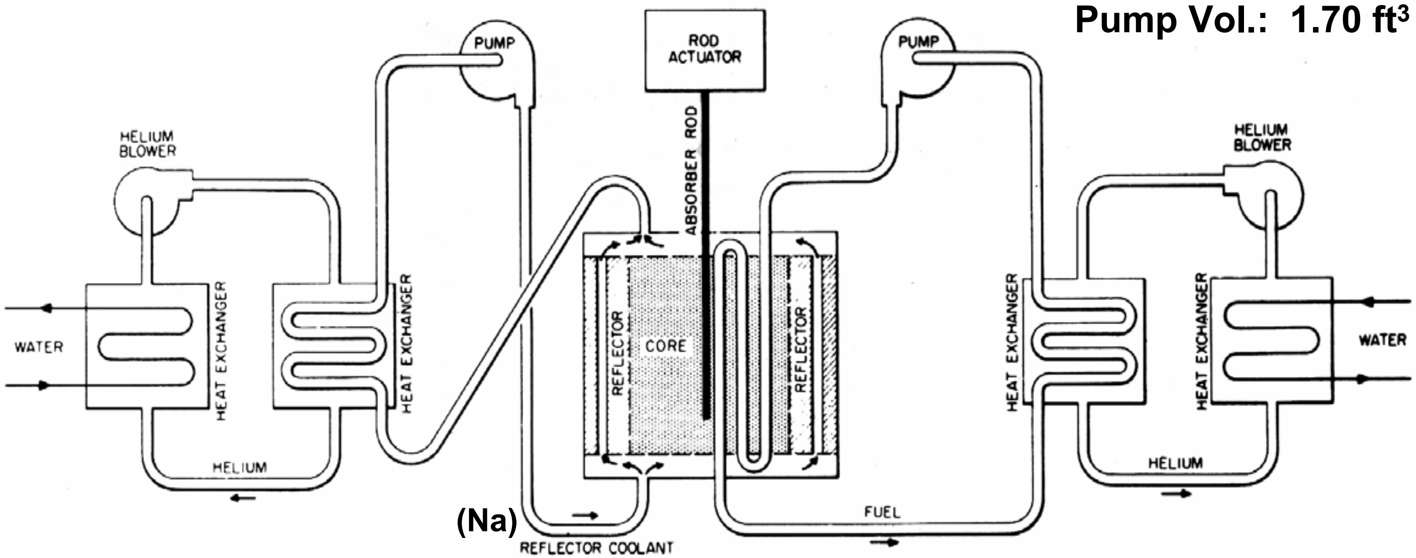


Molten Fluoride Salt Coolant Data

Aircraft Reactor Experiment (ARE) Successfully Demonstrated Molten Salt Reactor Technology in 1954

- Fuel: NaF-ZrF_4-UF_4 (53-41-6) (mole %)
- Sodium intermediate heat transfer loop
- Operated > 100Mw-h (2.5 MW(t)) for 2 months
- Max. fuel temp. 882°C; Material - Inconel
- Very large neg. temp. coeff (-6.1E-5)

Core Vol.: 1.37 ft^3
Loop Vol.: 3.60 ft^3
Pump Vol.: 1.70 ft^3



The Molten Salt Reactor Experiment Demonstrated Molten Salt Fueled Reactors And Intermediate Clean-Salt Heat-Transfer Loops

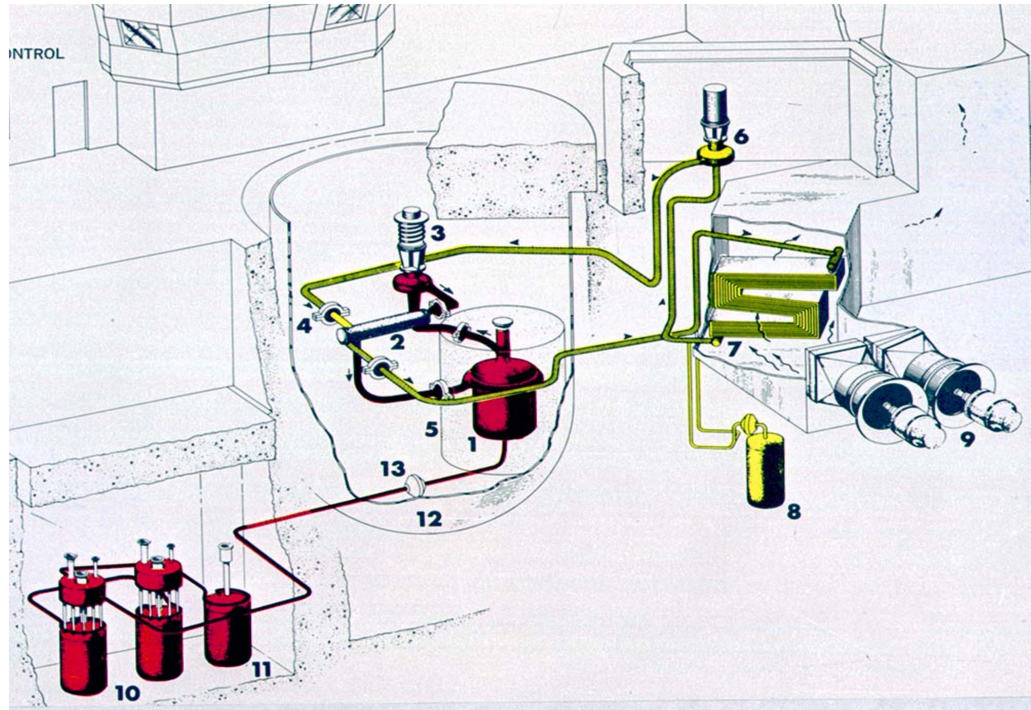
Hours critical	17,655
Circulating fuel loop time hours	21,788
Equiv. full power hrs w/ ^{235}U fuel	9,005
Equiv. full power hrs w/ ^{233}U fuel	4,167

U-235 fuel operation

- Critical — June 1, 1965
- Full power — May 23, 1966
- End operation — Mar 26, 1968

U-233 fuel operation

- Critical — Oct 2, 1968
- Full power — Jan 28, 1969
- Reactor shutdown — Dec 12, 1969



1. Reactor Vessel, 2. Heat Exchanger, 3. Fuel Pump, 4. Freeze Flange, 5. Thermal Shield, 6. Coolant Pump, 7. Radiator, 8. Coolant Drain Tank, 9. Fans, 10. Fuel Drain Tanks, 11. Flush Tank, 12. Containment Vessel, 13. Freeze Valve.

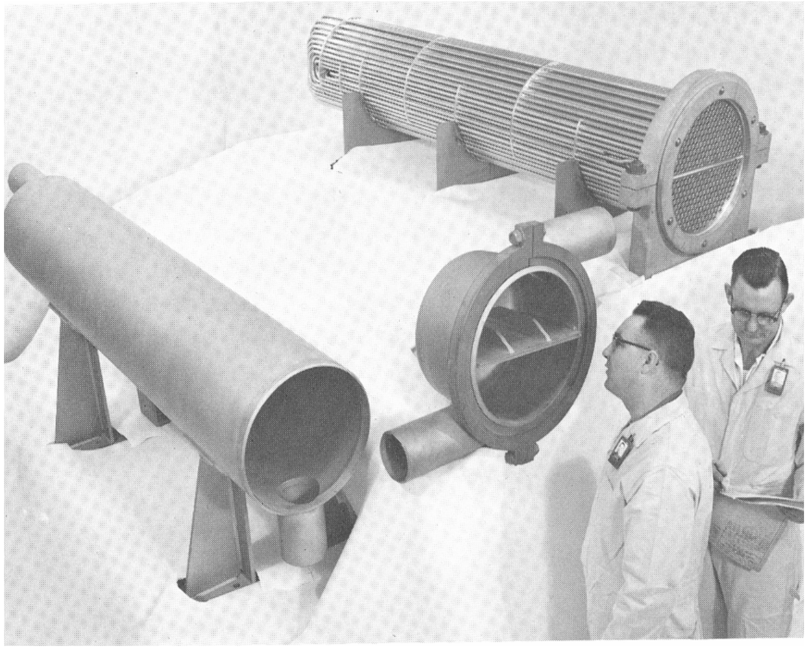
MSRE power = 8 MW(t)
Core volume <2 cubic meters

Molten Salt Reactor Experiment

Graphite Core

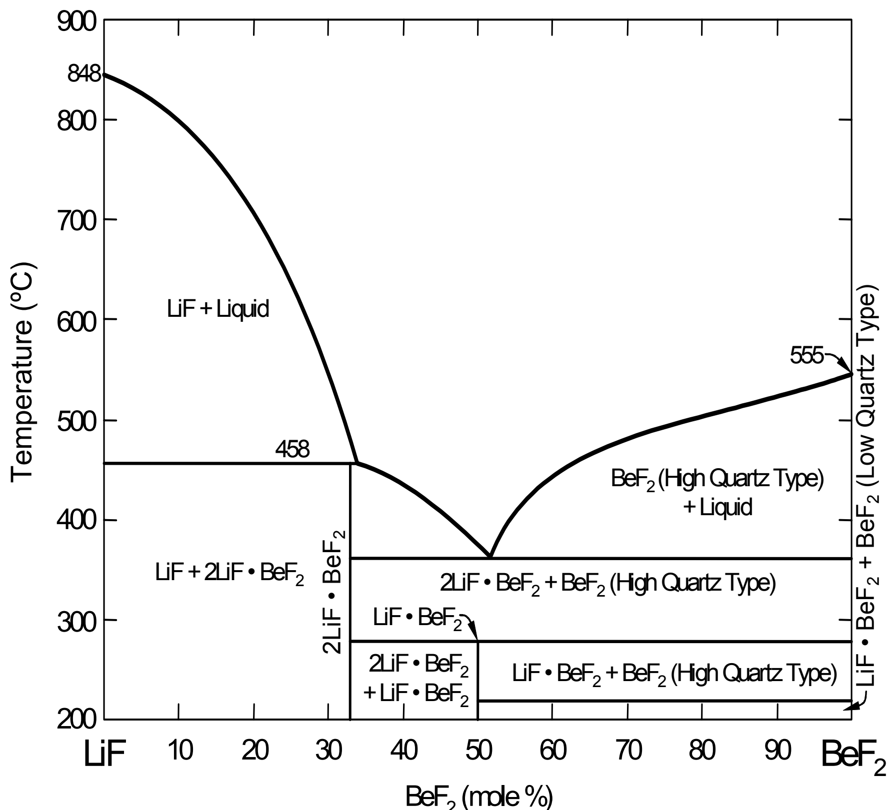
Intermediate Heat Exchanger

(Fuel Salt to Clean Salt)



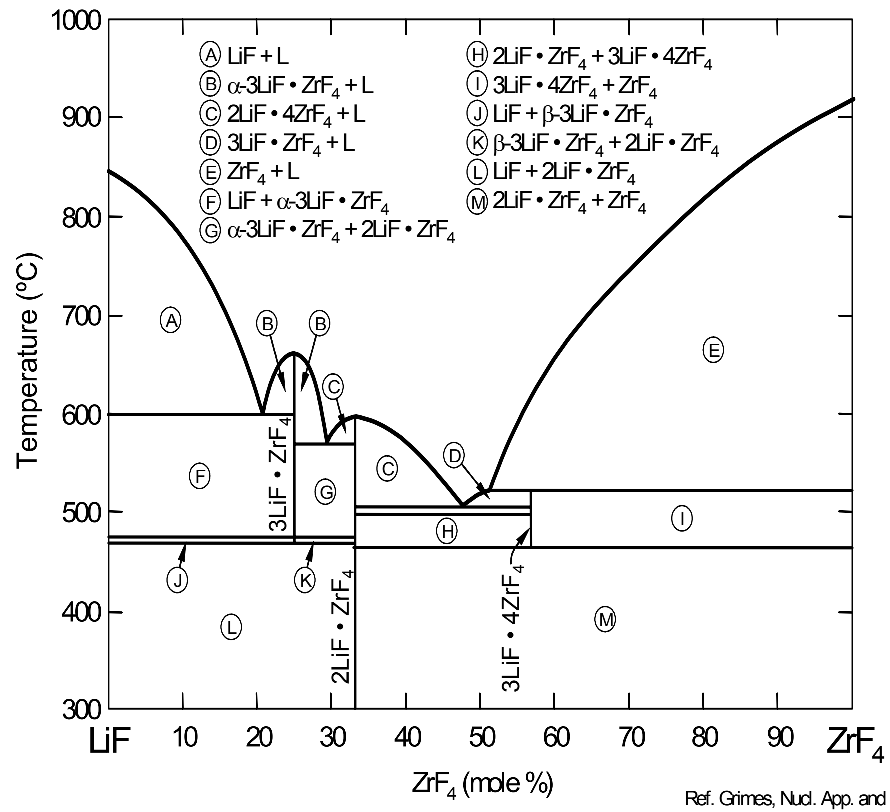
Multi-Component Salts Are Used To Improve Physical Properties, Such As Lowering Freezing Points

The System $LiF\text{-}BeF_2$



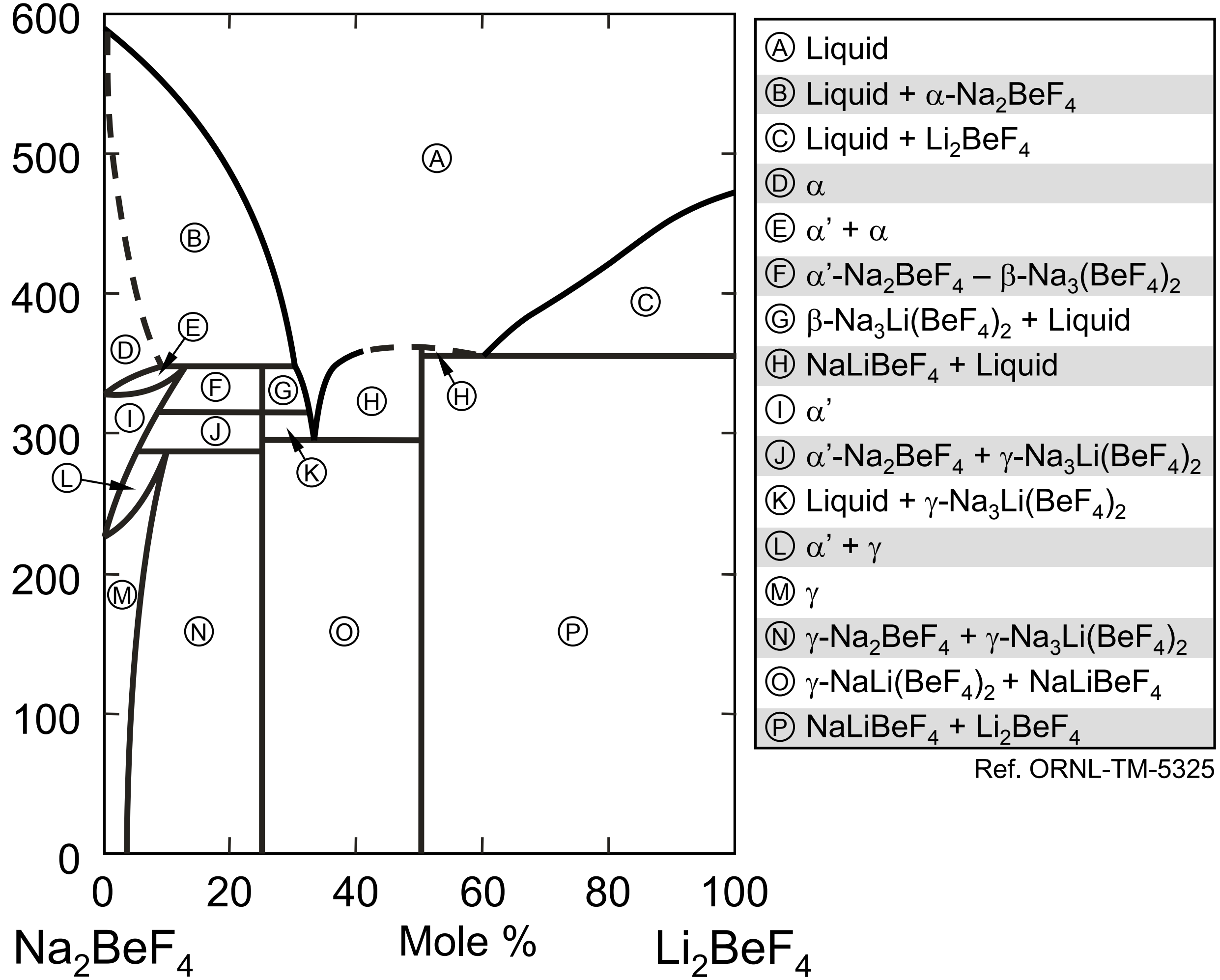
Ref. ORNL-TM-1853

The System $LiF\text{-}ZrF_4$



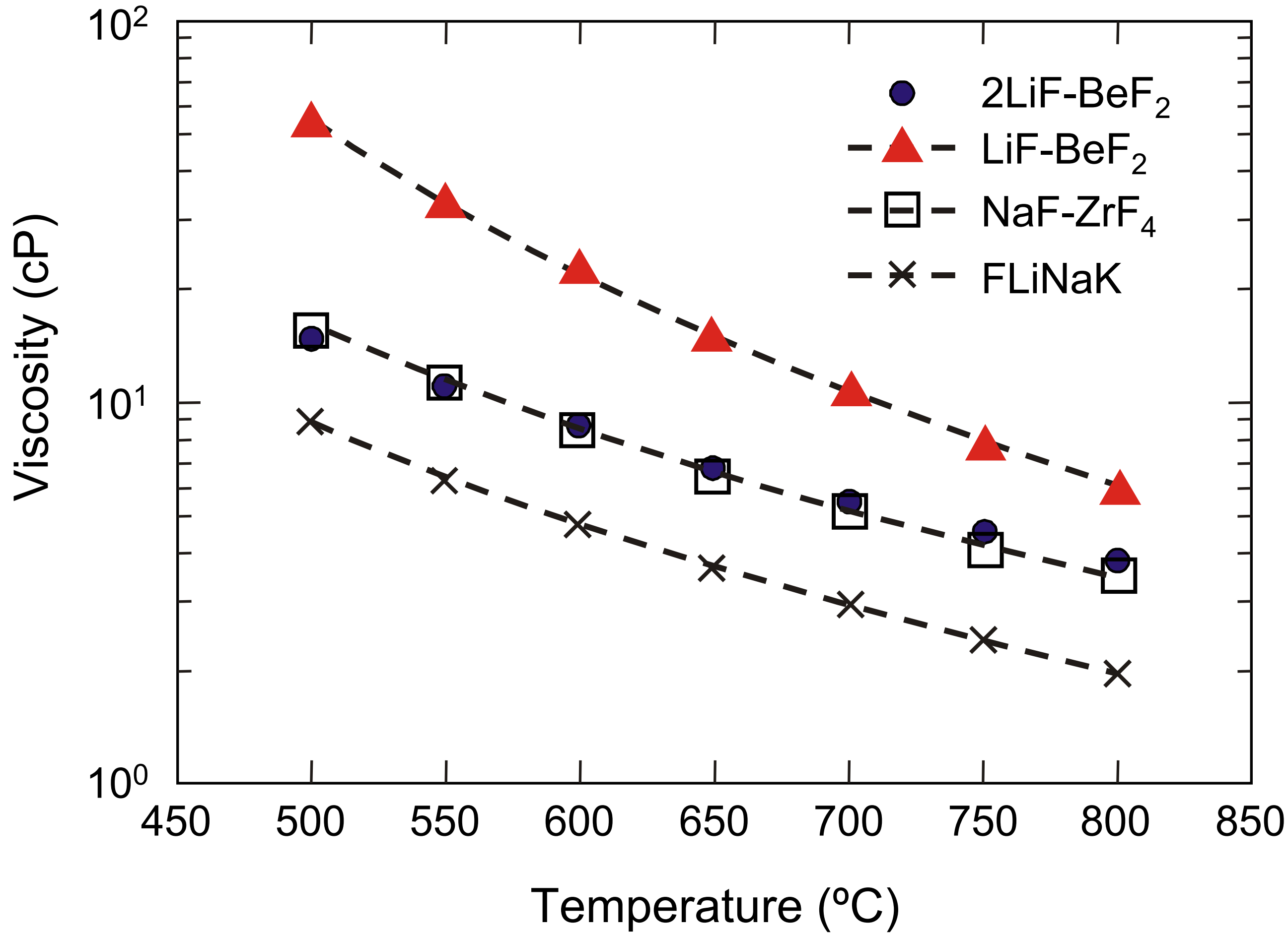
Ref. Grimes, Nucl. App. and Tech., Feb 1970

Freezing Points Can Be Lowered Using More Complex Salts (3 or 4 Components)



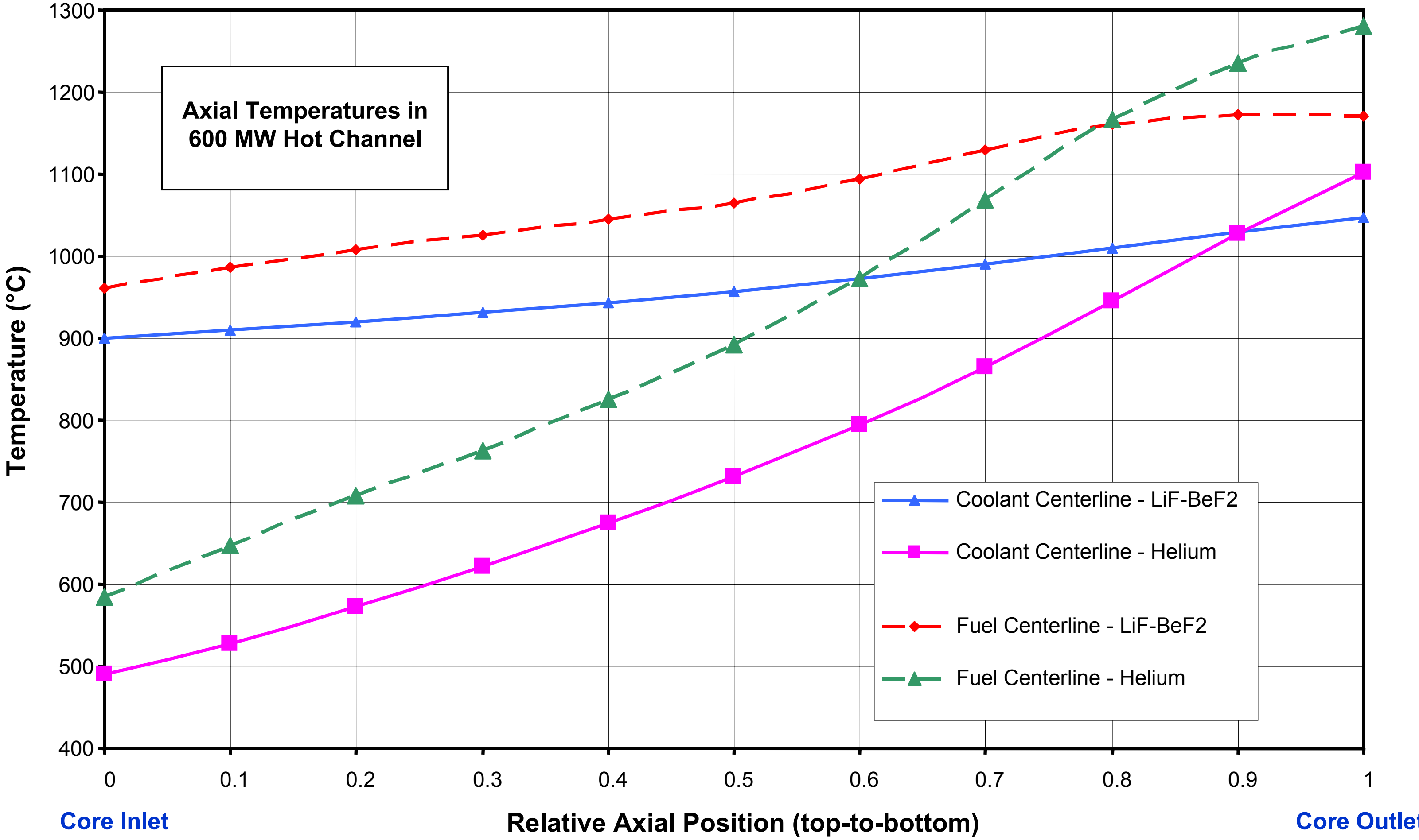
Ref. ORNL-TM-5325

Viscosity of Molten Fluoride Coolants



Liquid Transports Heat More Effectively Than Gas

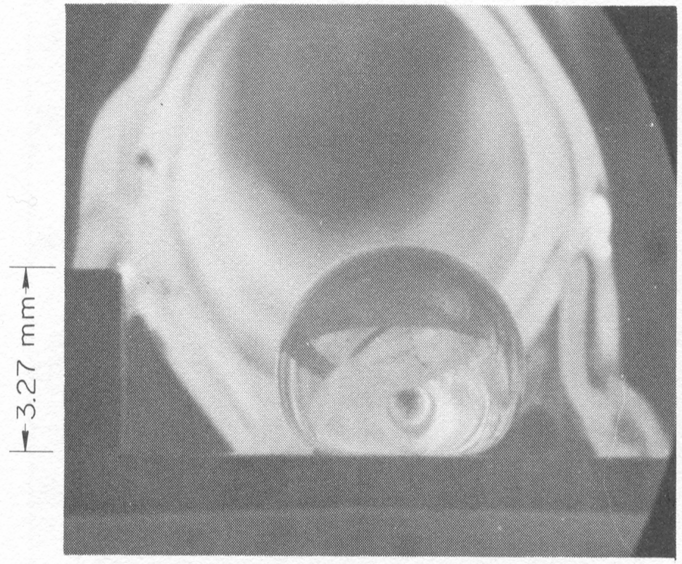
Smaller Temperature Differential Across Core



Materials Backup Data

Experience Shows Fluoride Salts Compatible With Carbon-Based Materials

- ***Graphite-molten salt temperature limits.*** **Graphite interactions with molten salt were investigated as part of the molten salt reactor program. Tests were conducted to 1400ºC with no interactions between salt and graphite (ORNL-4344). Postirradiation examination of graphite from the MSRE showed no interactions between salt and graphite.**
- ***Graphite-salt interactions.*** **The non-wetting behavior of the fluoride salts of interest implies that the molten salt will not penetrate small cracks in graphite and not contact the fuel matrix. Picture right shows the non-wetting behavior of 2LiF-BeF_2 on CGB graphite (contact angle ~ 150 degrees, see ORNL-3529, p. 125-129; see also ORNL-3591 p.38, ORNL-3626, p. 132; ORNL-3122, p. 93, and ORNL-4396, p. 210). ZrF_4-containing fluoride melts were also measured and found to be very non-wetting (~150 degrees) [J. of Nuclear Materials, Vol. 35, P. 87-93 (1970)].**



550°C 2 min AFTER MELTING

Experience Shows Fluoride Salts Compatible With Carbon-Based Materials

Evidence of lack of salt-graphite interaction from the post-irradiation examination of the MSRE (ORNL/TM-4174). A 1400ºC experiment was also conducted which showed salt-graphite compatibility (ORNL-4344 p112).

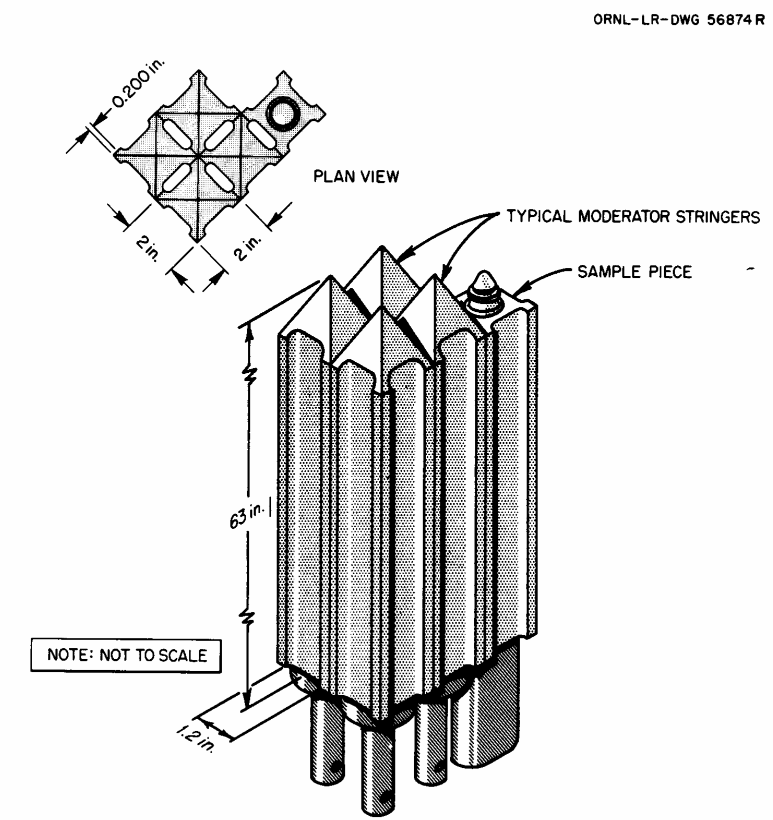


Fig. 4. Typical graphite stringer arrangement.

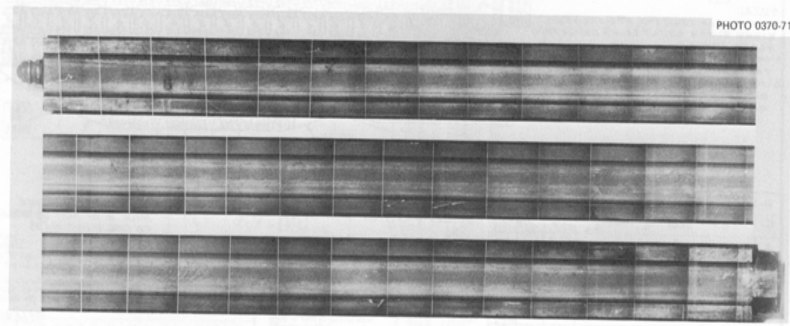


Fig. 6. One of the removable graphite moderator elements after operation in the MSRE, showing the fuel channel facing the center of the reactor. Machining marks are plainly visible in the fuel channel, showing excellent condition of the core block.

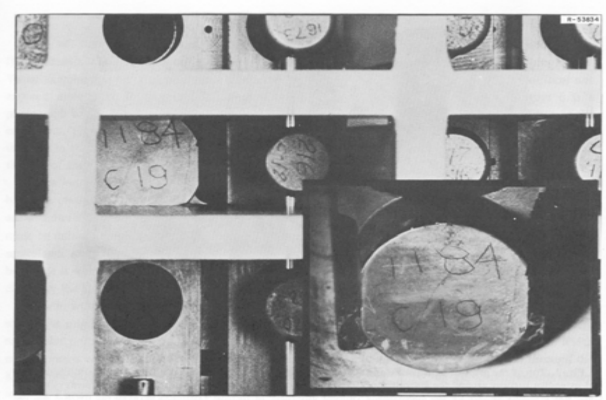


Fig. 8. Comparison of the bottom positioning stud on the graphite moderator element when initially installed in the MSRE and after operation. The large picture was made before the reactor was operational, and the small picture in the lower right corner was made in the hot cells after the moderator element had been at temperature for over three years.

Graphite & C-C Composites Are Generally Compatible with Molten Fluoride Salts

- Carbon components in reactor vessel will operate at 500 - 1100°C
  - Graphite core, reflector, and vessel insulation
  - C-C composite core supports, pump inlet, and control rods
- Considerations for service in AHTR
  - Salt compatibility, intercalation, wetting, strength (long-term and short-term), and irradiation-induced degradation and creep
- Extensive prior work has demonstrated compatibility and high resistance to intercalation and wetting with candidate salts
- Radiation-induced dimensional changes in graphite reactor vessel insulation must be accommodated
  - Provide torturous paths to vessel and/or periodic replacement
  - Cracking or failure of graphite insulation very unlikely at expected doses
- C-C composite heat exchangers should be explored as parallel path
  - Heat exchangers are highest metallic component performance risk

Other Graphite Issues Are Similar to Helium-Cooled, Prismatic VHTR with Known Solution Paths

Coated High-Temperature Alloys or Monolithic Alloys Will Likely Meet AHTR Needs

	Candidate Materials	Salt Corrosion Resistance	Air Corrosion Resist.	Long-Term Strength @ 1000°C	Highest Use T (°C)	Potential AHTR Component Usage*
Coated	Inconel 617	Needs Eval	Good	Very Good	1000	PM, P, V, HX
	VDM 602CA	Needs Eval	Good	Very Good	1000	P, V, HX
	Alloy 800 H	Needs Eval	Poor	Good	1000	P, HX
	Haynes 230	Needs Eval	Marginal	Good	900	P, HX
	Hastelloy X or XR	Needs Eval	Poor	Good	900	P, HX
	HP modified	Needs Eval	Good	Excellent	1100	V
Monolithic	Haynes 214	Very Good	Good	Good	1000	V, HX, CHX
	MA 956	Very Good	Good	Good	?	HX, CHX
	MA 754	Very Good	Good	Good	?	HX, CHX
	Cast Ni Superalloys	Very Good	Good	Good	?	PM
	Candidate Materials	Metallurgical Stability	Irrad. Resistanc	Fabricability	Alloy Maturity	Codified
Coated	Inconel 617	Good	Good	Good	High	Sec VIII
	VDM 602CA	Good	N/A	Good	Medium	Sec VIII
	Alloy 800 H	Good	N/A	Good	High	Sec I, III, VIII
	Haynes 230	Good	N/A	Fair-Good	High	Sec I, VIII
	Hastelloy X or XR	Good	N/A	Good	High	Sect I, VIII
	HP modified	Good	N/A	Cast Only	High	API
Monolithic	Haynes 214	Fair-Poor	N/A	Poor-Fair	Low	No
	MA 956	Good	N/A	Poor-Fair	Low	No
	MA 754	Good	N/A	Poor-Fair	Low	No
	Cast Ni Superalloys	Good	Adequate	Cast Only	High	No

* Pump (PM), Piping (P), Valves (V), Heat Exch's (HX), Compact HX (CHX)

Coated F-M or Stainless Steels or Monolithic Alloys Will Likely Meet AHTR Reactor Vessel Needs

	Candidate Materials	Salt Corrosion Resistance	Air Corrosion Resistance	Long-Term Strength @ 500°C	Highest Usage Temp (°C)
Coated	9Cr-1MoV	Poor	Good	Very Good	650
	2 1/4 Cr-1Mo	Poor	Good	Good	650
	304	Poor	Good	Very Good	815
	316	Poor	Good	Very Good	815
	347	Poor	Good	Very Good	815
	Alloy 800H or HT	Poor-Fair	Good	Very Good	980
Monolithic	Hastelloy N	Excellent	Good	Very Good	730
	Haynes 242	Very Good	Good	Very Good	540
	Alloy 800H or HT	Poor-Fair	Good	Very Good	980

	Candidate Materials	Metallurgical Stability	Irradiation Resistance	Fabricability	Maturity	Codified
Coated	9Cr-1MoV	Fair	Good	Good	High	Sec III, VIII
	304	Good	Good	Good	High	Sec III, VIII
	316	Good	Good	Good	High	Sec III, VIII
	347	Good	Good	Good	High	Sec III, VIII
	Alloy 800H or HT	Good	Good	Good	High	Sec I, III, VIII
Monolithic	Hastelloy N	Good	Good	Good	High	Sec III*, VIII
	Haynes 242	Good	Adequate	Good	Low	Sec VIII
	Alloy 800H or HT	Good	Good	Good	High	Sec III, VIII

*Existing Code Case

Selected Metals Are Compatible With Clean Molten Fluoride Salts, but There Is a Lack of Very-High-Temperature Data

- Corrosion control by thermodynamics
  - Metals are noble with respect to salt
  - Same approach as used with sodium coolants
  - Coolant chemistry control to maintain reducing conditions can extend material performance
- Hastelloy-N code qualified (to 750°C)
  - Higher-temperature materials required for this application
  - Candidate materials similar to those proposed for the helium-cooled VHTR
  - However, the amount of higher-temperature corrosion test data in corrosion test flow loops is limited

Facility Design Data

Proposed AHTR Facility Layouts Are Similar to Sodium-Cooled Fast Reactors (1 of 2)

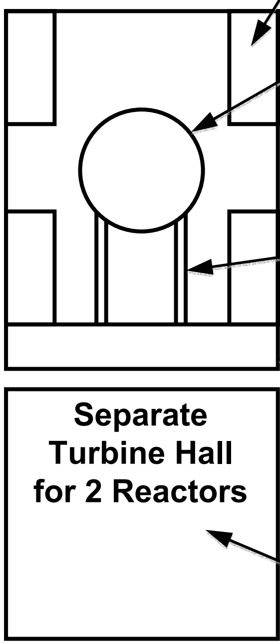
- *Similarities*. Sodium-cooled fast reactors and the AHTR are both low-pressure high-temperature reactors. Consequently, the general plant layout and much of the licensing basis will be very similar to those of sodium-cooled reactors. As a starting point, the AHTR facility design follows that of the General Electric S-PRISM. S-PRISM is a modular reactor with each module producing 1000 MW(t) output and 380 MW(e). It is the last sodium-cooled reactor that was designed in the United States.
- *S-PRISM design*. The design goals were for a passively safe economic breeder reactor. The economic optimization of the S-PRISM indicated that larger modules were more economical. The modular size limit was controlled by the ability to passively remove decay heat from the reactor vessel. Decay heat removal capability depended upon vessel size. A series of detailed engineering studies resulted in a vessel about 9.2 meters in diameter. This was defined as the largest practical size given various engineering, cost, and fabrication limitations. The large, low-pressure vessel had sufficient space for the reactor core, intermediate heat exchangers, and spent fuel storage.
- *AHTR design basis*. As a starting point, the S-PRISM facility design was used as a basis for the AHTR. This included using the same size reactor vessel. Because the AHTR is also a low-pressure liquid-cooled reactor, it is a reasonable starting assumption to assume the same fundamental limitations in facility and vessel design.
- *AHTR Differences*. There are several differences in plant layout.
  - The intermediate heat exchangers and SNF storage are removed from the reactor vessel to provide space for the larger AHTR core. The intermediate heat exchangers are moved to the compartment that in S-PRISM contains the sodium-water heat exchangers.
  - The molten salt-gas (nitrogen or helium) heat exchangers in the secondary heat transfer loop are on the turbine floor. This is required for an efficient Brayton cycle. The pressure drops in gas systems are very large compared with molten salt pressure drops. As a consequence, the molten salt gas heat exchangers must be next to the Brayton cycle turbines
  - The larger AHTR results in a single Brayton cycle per reactor rather than the S-PRISM system, where two reactors provide thermal energy to a single Rankine (steam) plant.

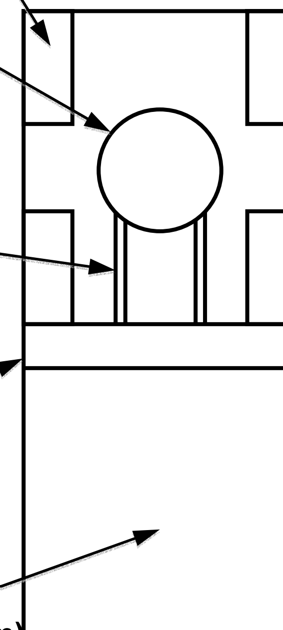
Proposed AHTR Facility Layouts Are Similar to Sodium-Cooled Fast Reactors (2 of 2)

- ***Containment.*** **The AHTR containment building requirements are less than S-PRISM or a helium-cooled VHTR.**
  - **Pressurization. The chemical reactions of sodium with water and the high-pressure associated with helium-cooled reactors create gases that can pressurize containments and other structures. These pressure mechanisms do not exist for the AHTR.**
  - **Radionuclide release. Molten salts dissolve most fission products (including cesium and iodine) to very high temperatures. This creates an additional barrier to fission product release that does not exist with other reactors.**
- ***Balance of Plant.*** **The AHTR, like S-PRISM, uses an indirect power cycle. This has several implication relative to helium-cooled reactors.**
  - **High-quality industrial (not nuclear) standards, construction, and maintenance may be used for the power cycle.**
  - **Fuel quality. The absolute requirements for fuel reliability are lower than for direct cycle power plants. Radionuclides from leaky fuel dissolve in the coolant and are removed by the coolant cleanup system.**
- ***Refueling*****. The refueling, operation, and maintenance of the AHTR will have many similarities to sodium- and lead-cooled fast reactors. The AHTR uses separate SNF storage, it does not use the reactor vessel for SNF storage. In some respects, the AHTR operations will be simpler because (1) the salt is transparent and thus allows camera views of the reactor core and (2) the salt is less chemically reactive than sodium. However, the temperatures will be higher and will require more careful design of systems to avoid undesired salt freezing.**
- ***Licensing*****. Very little work has been done to develop a regulatory framework for the AHTR. The NRC review of the conceptual design of S-PRISM (February 1994) provides the initial basis for the regulatory consideration of a pool-type reactor. The use of the same fuel as the fuel used in gas-cooled reactors implies that many of the licensing interactions associated with gas-cooled reactors are directly applicable. However, the licensing issues with fuel performance are less because of (1) use of a low-pressure system, (2) a coolant that dissolves fission products, and (3) an indirect power cycle that further isolates the reactor core from the environment.**
- ***Inspection.*** **Molten salts are transparent. This is a major advantage over liquid-metal reactors where inspection has been a major issue.**

A 2400-MW(t)* AHTR Nuclear Island Size May Ultimately Be Similar to A 1000-MW(t) S-PRISM

(Consequence of Higher Temperatures and Different Coolant Properties)





S-PRISM (Sodium)	AHTR (Molten Salt)
Passive Decay Heat Removal	
• Same Vessel Size • Peak Vessel Temp ~550°C	• Same Vessel Size • Peak Vessel Temp ~750°C
Same Vessel Size	
• Reactor Core • Sodium/Sodium Heat Exchanger	• Reactor Core
Similar Equipment Size (Volumetric Heat Capacity: Molten Salt > Sodium)	
Heat Exchanger Compartment	
• Sodium/Water Heat Exchanger	• Molten Salt/ Molten Salt Heat Exchanger
Turbine Hall	
• Rankine (Steam)	• Brayton (Nitrogen or Helium) – Gas/Molten Salt Heat Exchanger – Higher Efficiency (Higher Temperature)

*AHTR 1200 MW(e): S-Prism 380 MW(e)

AHTR Nuclear Island (2400 MW(t)/>1200 MW(e)) Is Similar In Size To S-PRISM (1000 MW(t)/380 MW(e))

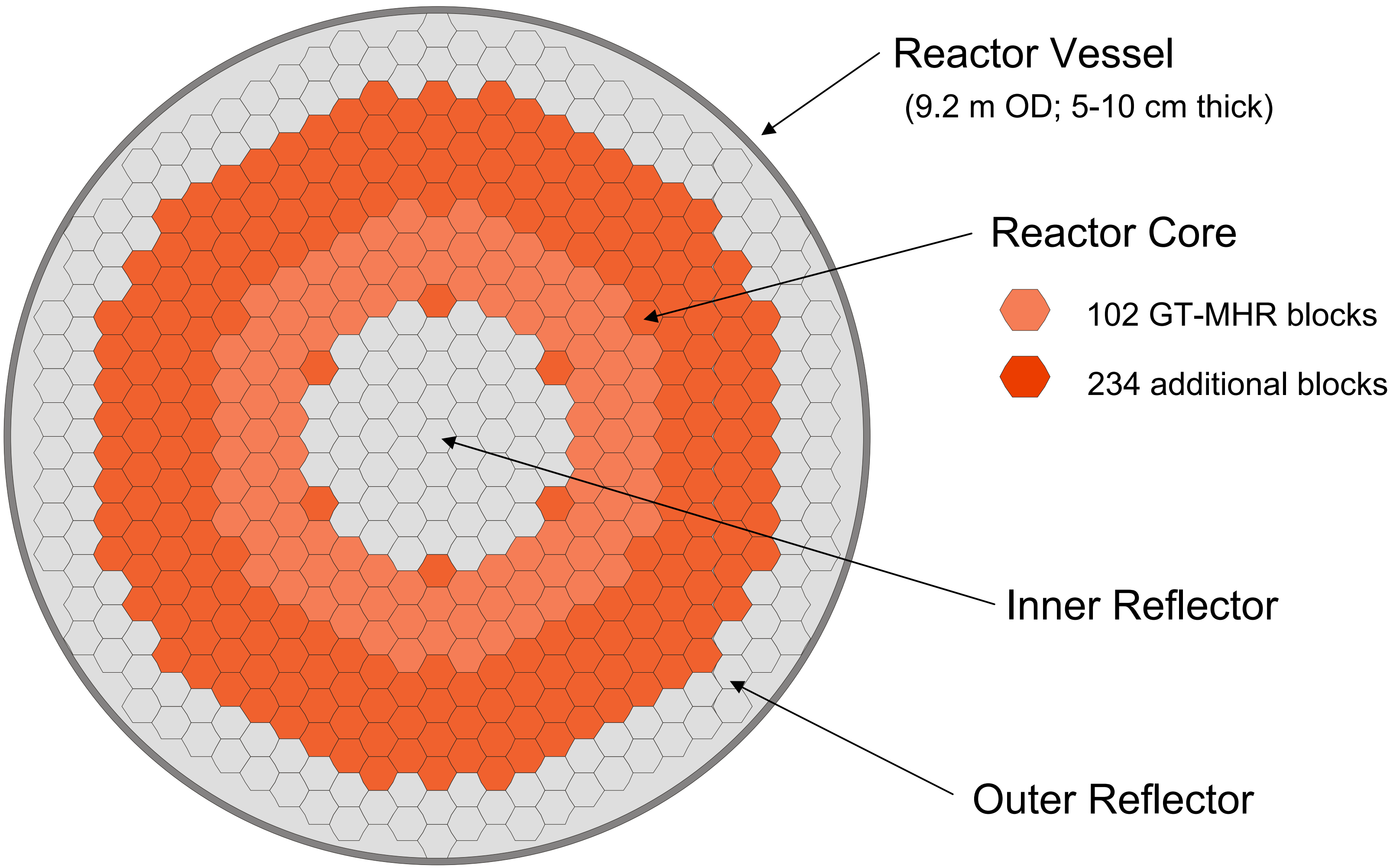
- ***AHTR Sizing.*** **The initial assumption was that the AHTR vessel would be the same size as the General Electric S-PRISM vessel. Both reactors are low-temperature high-pressure machines. General Electric did a series of trade studies that indicated that the best economics resulted in maximizing the reactor size because this maximized the reactor size possible with passive safety. Assuming that the AHTR power density will be similar to proposed gas-cooled coated-particle fuel reactors, the maximum power level is ~2400 MW(t). Using similar power densities as a gas-cooled reactor has several implications.**
  - **Same technology. The same fuels can be used for the AHTR as for gas-cooled reactors.**
  - **Similar vessel heat capacity. The reactor vessel heat capacity is about the same per MW(t) as a gas-cooled reactor. This implies similar slow vessel heatup rates in an accident.**
  - **Lower peak fuel temperatures. Liquids are better coolants than gases. With similar power densities, the fuel temperatures will be less than those in gas-cooled reactors.**
- ***Technical viability*****. The nuclear island size for the 1000 MW(t) S-PRISM and 2400 MW(t) AHTR are about equal because of several factors**
  - **Vessel volume. The larger core (see earlier) is possible because S-PRISM includes SNF storage and the intermediate heat exchangers in the reactor vessel while these are moved out of the vessel in the AHTR design.**
  - **Equipment size. Pipes, pumps, and valves are similar in size because the volumetric heat capacity of molten salts is several times greater than sodium. Volumetric heat capacity sizes much of the equipment.**
  - **Decay heat removal. The AHTR operates at higher temperatures. The decay heat removal system performance is a strong function of temperature.**
  - **Electricity production. The higher electricity production is a consequence of higher total power levels and higher temperatures (more efficient generation of electricity**
- ***Other size considerations*****. There are several other factors not related to S-PRISM that makes a 2400 MW(t) reactor of interest**
  - **Electricity output. The reactor electrical output is similar to other large nuclear reactors used for power production**
  - **Hydrogen. The largest hydrogen plant under construction using natural gas as an energy source will produce 300 million ft^3/day. For a high-temperature reactor to produce an equivalent quantity of hydrogen, the energy output must be about 2400 MW(t).**
  - **Beyond-design-basis accidents. Preliminary analysis (further in presentation) indicates that this size of reactor may be able to withstand beyond-design-basis accidents (vessel failure) without massive fuel failure**

Beyond-Design-Basis Accident Avoids Radionuclide Release By Multiple Mechanisms

- ***Dissolution of radionuclides in molten salt*. All actinides and most fission products (including cesium and iodine) are highly soluble in fluoride molten salts to very high temperatures. This was the basis for the molten salt reactor where the uranium and plutonium was dissolved in the coolant. The only exceptions are the noble gases (primarily krypton and xenon) and the noble metal fission products that can plate out on surfaces. As a consequence of this behavior, as long as the solid fuel is in the salt, actinides and fission products from failed fuel will dissolve in the salt and not escape the reactor**
- ***Salt isolation of SNF from air.* One of the safety issues associated with graphite fuels is air ingress and oxidation of the fuel. As long as the fuel is covered by the coolant, air can not reach the fuel**
- ***No pressurization of containment.* Molten salts do not pressurize containment under accident conditions. This avoids a major energy source for dispersal of actinides and fission products. Gas cooled reactors typically include vented containments to allow escape of the helium if the reactor system depressurizes.**
- ***Beyond-design-basis-accidents*. In a beyond design basis accident with vessel failure, the molten salt floods the bottom of the silo with molten salt while keeping the reactor core covered with coolant. The molten salt may efficiently allow heat transfer to the silo while maintaining peak temperatures below those of massive fuel failure. See: C. W. Forsberg and Per F. Peterson, "Making Core Melt Accidents Impossible In a Large 2400 MW(t) Reactor, Global 2003", *Embedded Topical within 2003 American Nuclear Society Winter Meeting*, Nov. 16-20, 2003, New Orleans, Louisiana. However, significant work is required before this safety strategy can be considered proven (The first three safety mechanisms have significant data to support the conclusions).**

Reactor Core Design

Detailed Core View



Void Coefficient vs. Salt Choice – SNL Model

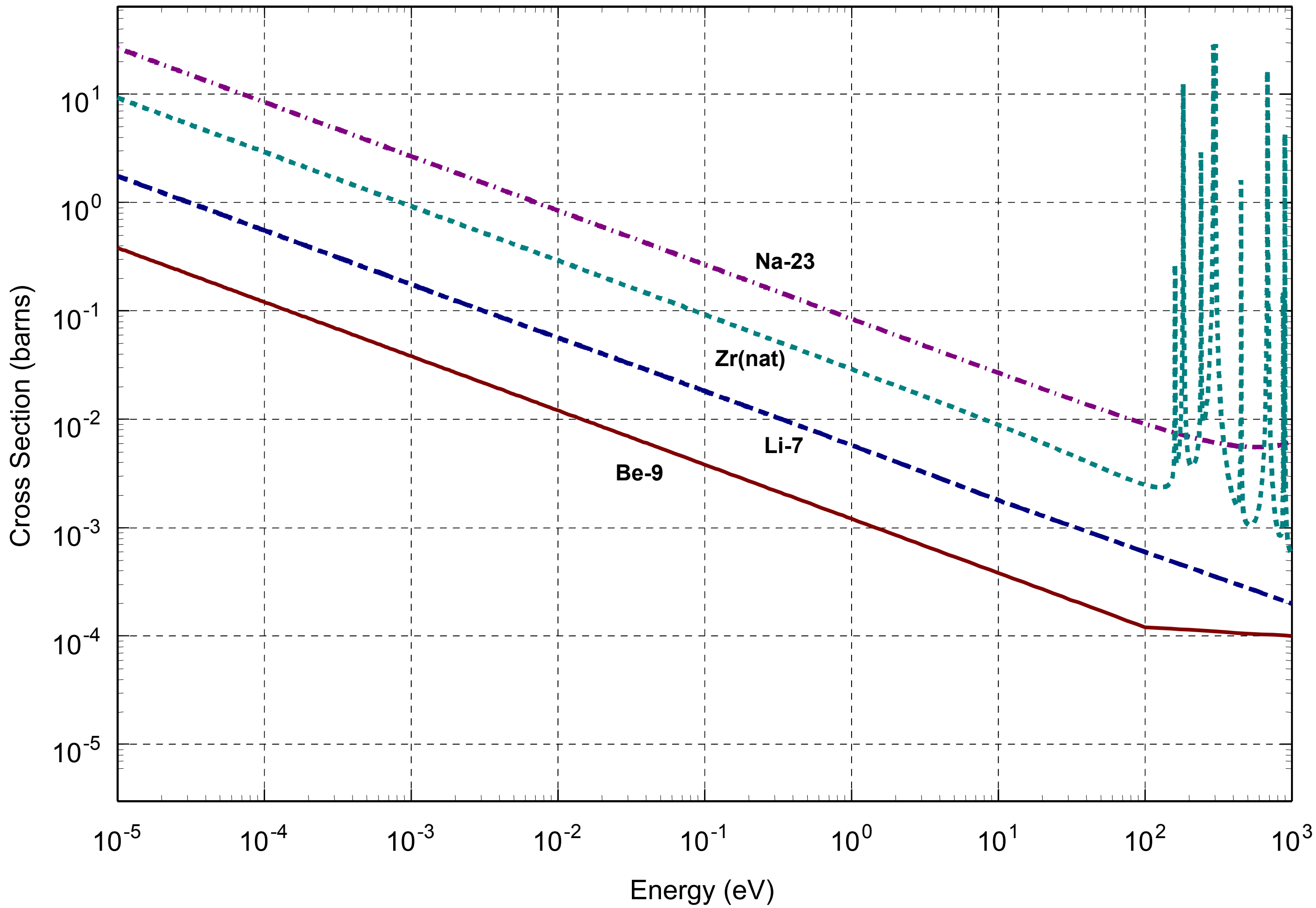
Salt	Total Void Reactivity Effect ($)
BeF_2	-1.46
LiF/BeF_2 (66/34)	-0.47
MgF_2/BeF_2 (50/50)	-0.49
-------------------------	-------------------------
LiF (Li-7)	+0.16
ZrF_4/BeF_2 (50/50)	+0.43
ZrF_4/LiF (52/48)	+1.25
-------------------------	-------------------------
NaF/BeF_2 (57/43)	+1.82
ZrF_4	+1.41
NaF/ZrF_4 (25/75)	+1.88
NaF/ZrF_4 (50/50)	+2.64
NaF/ZrF_4 (75/25)	+3.82
NaF	+7.05

- Example for 6.6% coolant volume fraction and complete core voiding
- Lower mass elements with low absorption cross sections can yield negative feedback
- Positive effects are reactivity limited

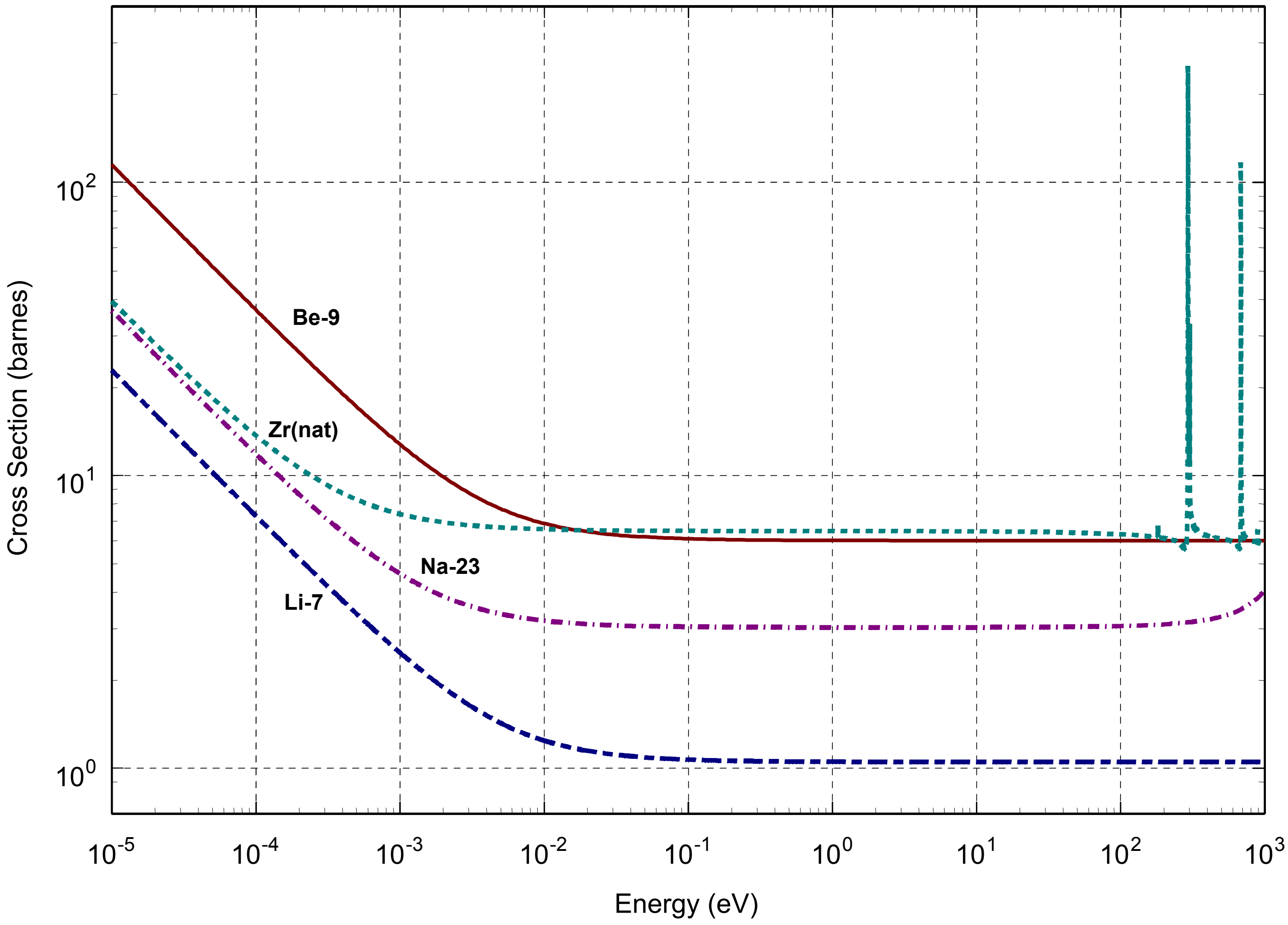
<u>Ranking (best to worst)</u>

Be, Li-7, Mg, Zr, Na

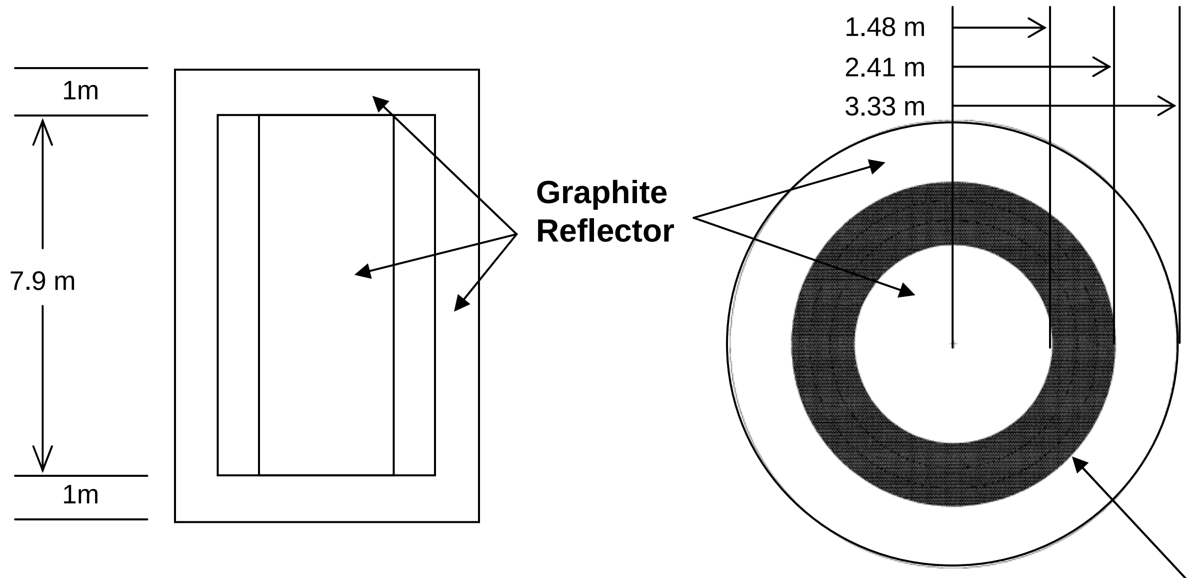
Void Coefficient Sensitive To Capture Cross Section of Salt Constituents

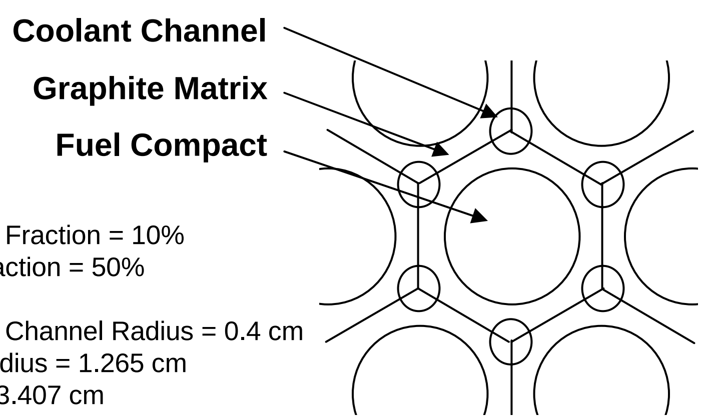


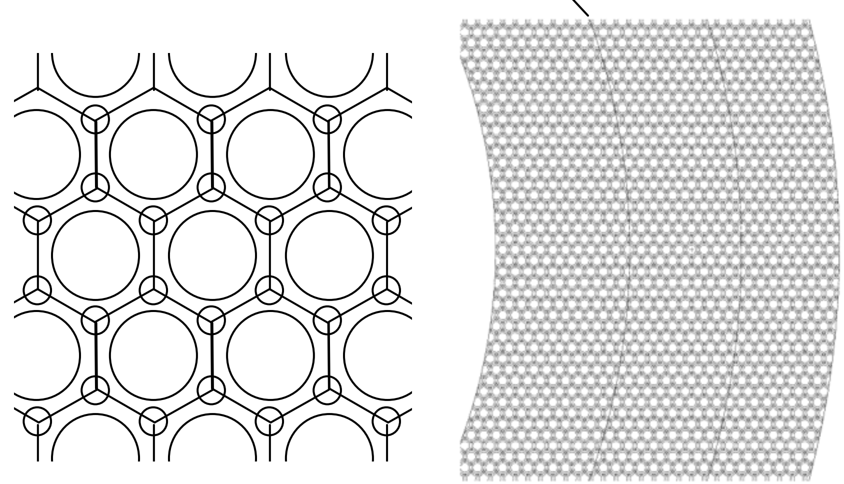
Void Coefficient Also Sensitive to Scattering Cross Section of Salt Constituents



Geometry for SNL MCNP Calculations







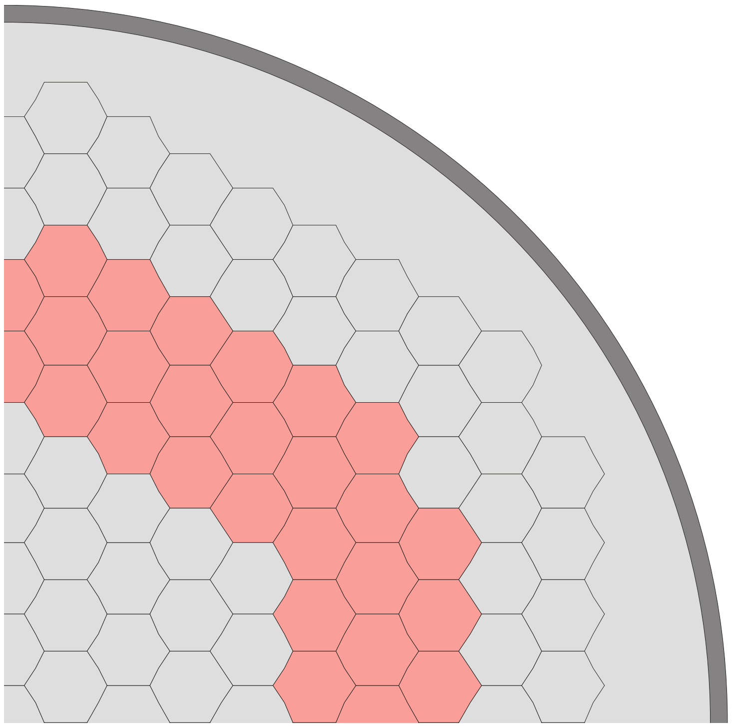
Coolant Fraction = 10%
Fuel Fraction = 50%

Coolant Channel Radius = 0.4 cm
Fuel Radius = 1.265 cm
Pitch = 3.407 cm

Fuel Particle Packing Fraction = 0.3

Impact of Burnable Poisons and ^{7}Li Purity on Void Coefficient – ORNL Model

BP Loading (grams of Er per block)	**Lithium-6 Enrichment** (atom-%)	**Void Coeff. for full core voiding** ($)
70	**0.01**	**1.30**
140	**0.01**	**1.12**
210	**0.01**	**0.98**
210	**0.0**	**0.42**

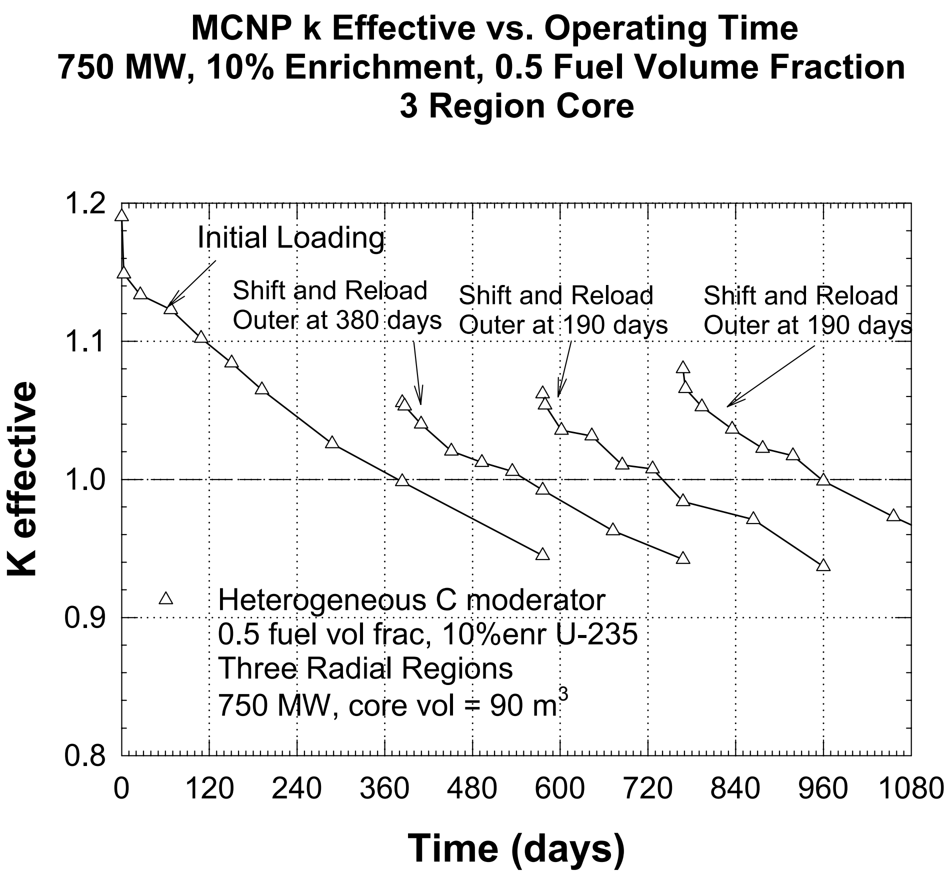


- **66%LiF-34%BeF_2 Salt**
- **1 mol% VF_3 Buffer**
- **102-column core (600MW)**
- **900 °C coolant; 1200 °C fuel**
- **14 wt% ^{235}U**

Conclusions on Void Coefficient

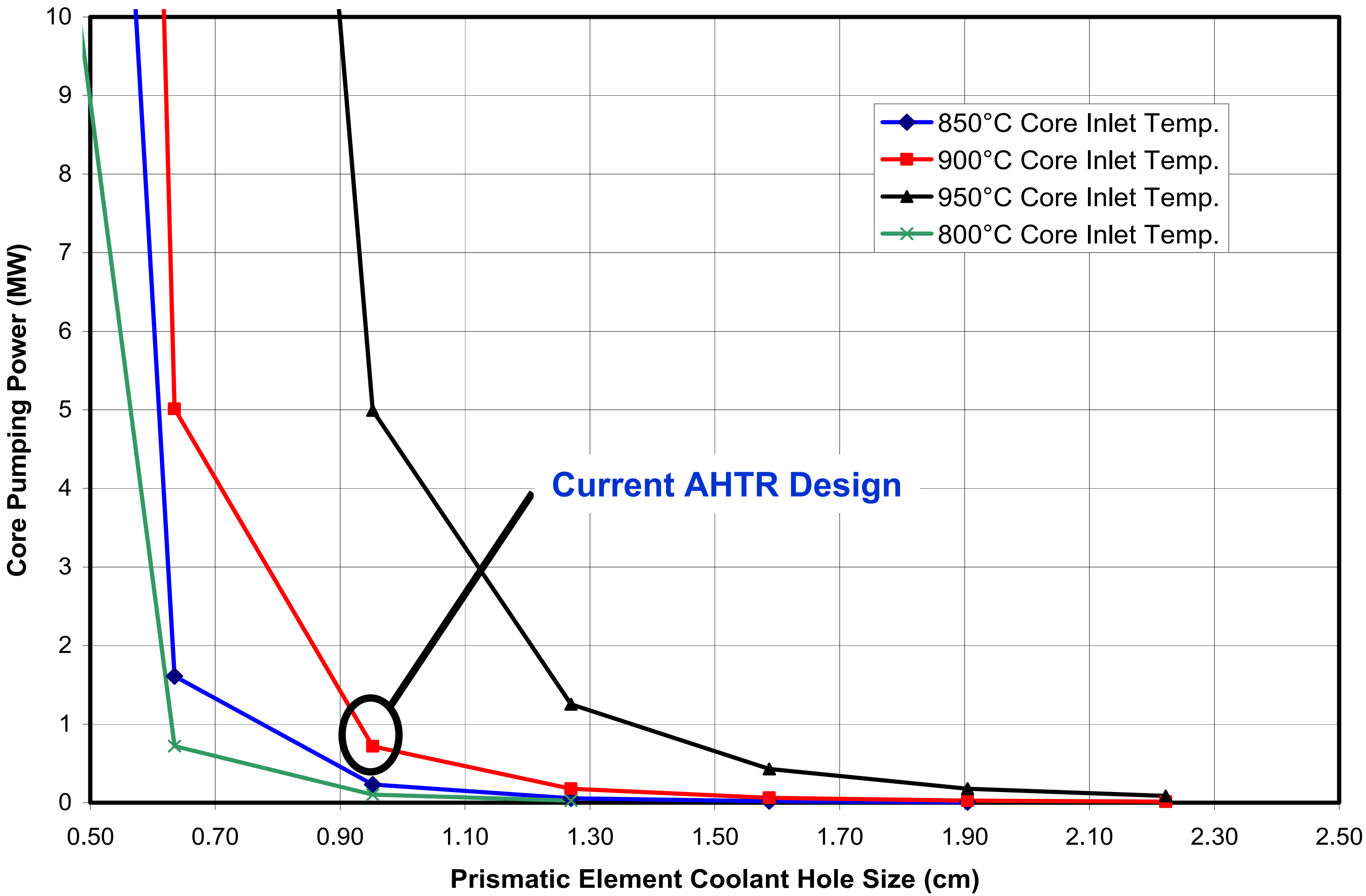
- **Magnitude of coolant void coefficient (CVC) for LiF-BeF_2 decreases with increasing uranium loading and increasing burnable poison loading**
- **Magnitude of CVC depends on the neutron spectrum – it decreases with increasing U/C ratio**
- **Magnitude of CVC for LiF-BeF_2 is very dependent on the ^{7}Li purity in the salt**
- **Magnitude of CVC increases rapidly with increasing coolant hole diameter – higher relative coolant volume in the core**
- **Magnitude of CVC can be decreased by making the neutron spectrum of the coolant channels harder – higher fuel loading, higher burnable poison loading, poisoning the graphite blocks or replacing some of the graphite in the core with carbides, incorporate the coolant channels at the center of the fuel channels.**
- **Need substantial neutronics analysis to evaluate options for reducing or zeroing the CVC**

Burnup Example for 8.3 W/cc – 90m^3 Core

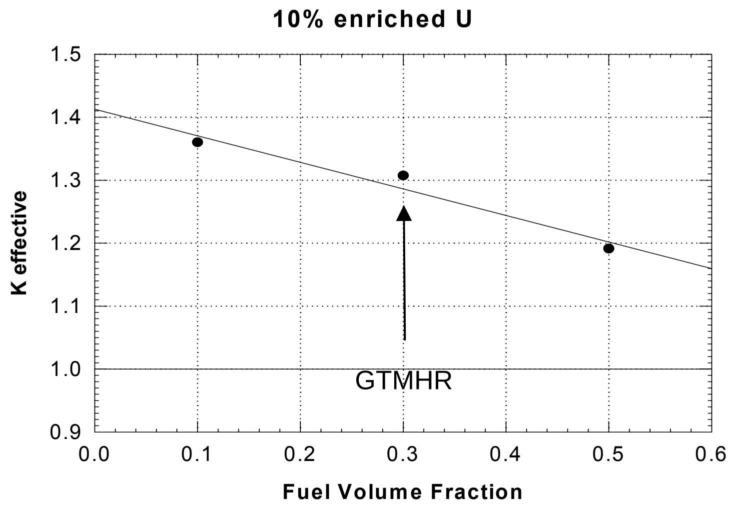


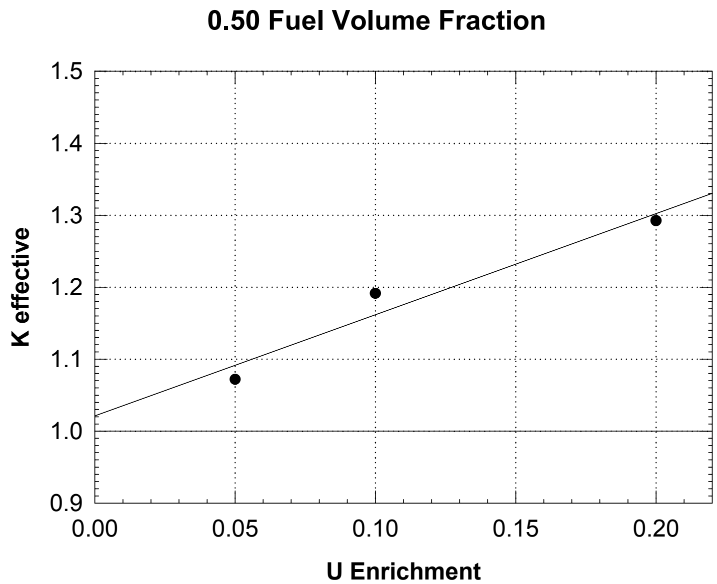
- Core life considerations similar to GTMHR
- 8.3 W/cm^3, core life is approximately 580 days (10% enrich, 0.5 fuel volume fraction
- Burnup similar to GT-MHR

Pumping Power Depends Heavily On Core ΔT and Coolant Channel Diameter



Neutronics for AHTR Are Similar to GTMHR





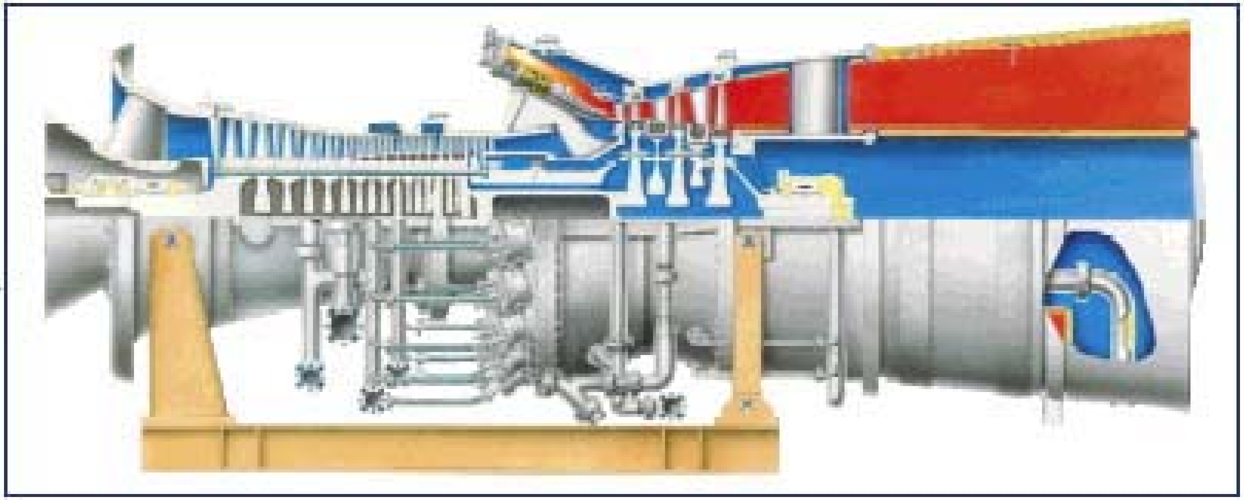
- Excess reactivity similar for given core loading
- Lower coolant volume fraction
- Neutron lifetime ~1ms
- k_{eff} increases with higher moderator to fuel ratio (undermoderated in design region)
- Similar fuel burnup and fuel cycle options
- Large negative temperature coefficient due to Doppler effects ~ -$0.01/K

Void Reactivity Worth Molten Salt/Graphite/Fuel Matrix Reactor
Core Radius = 241cm outer, 148cm inner, Height = 793cm
Volume Fraction of Flibe Salt = 0.065
Inner Region Graphite, Outer Reflector 92cm Graphite
Axial Reflector = 100 cm Graphite
Temperature = 900 ^oC
0.3 Particle Packing Fraction

Energy Conversion Data

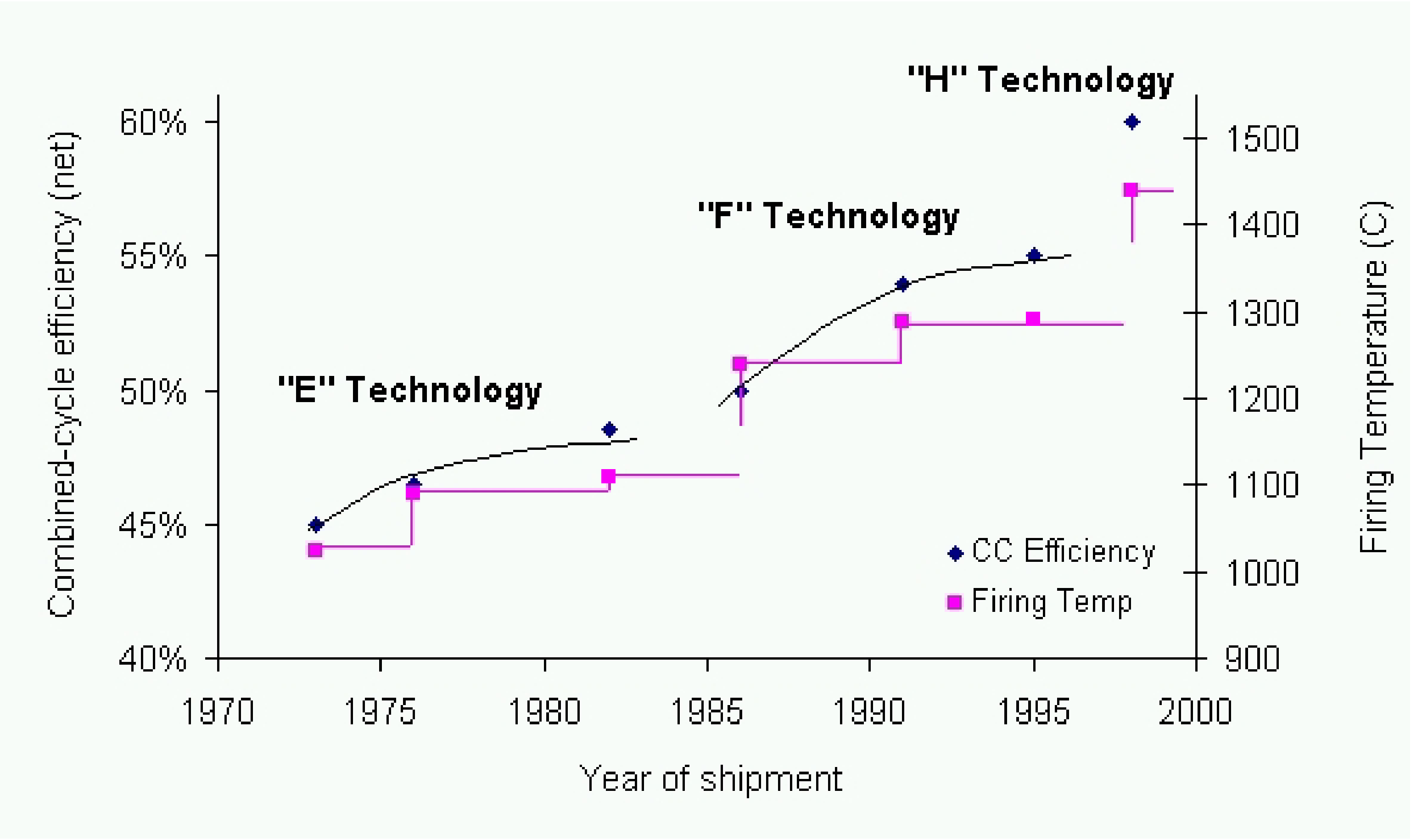
High Temperature Turbines

GE Power Systems MS7001FB
Introduced in November 1999; 9 purchased for New York plant
Designed for high efficiency, low life-cycle cost power plants
Uses single crystal materials developed for GE's jet engines
2500° F-class firing temperature (1644 K, 1371 ^{o}C)



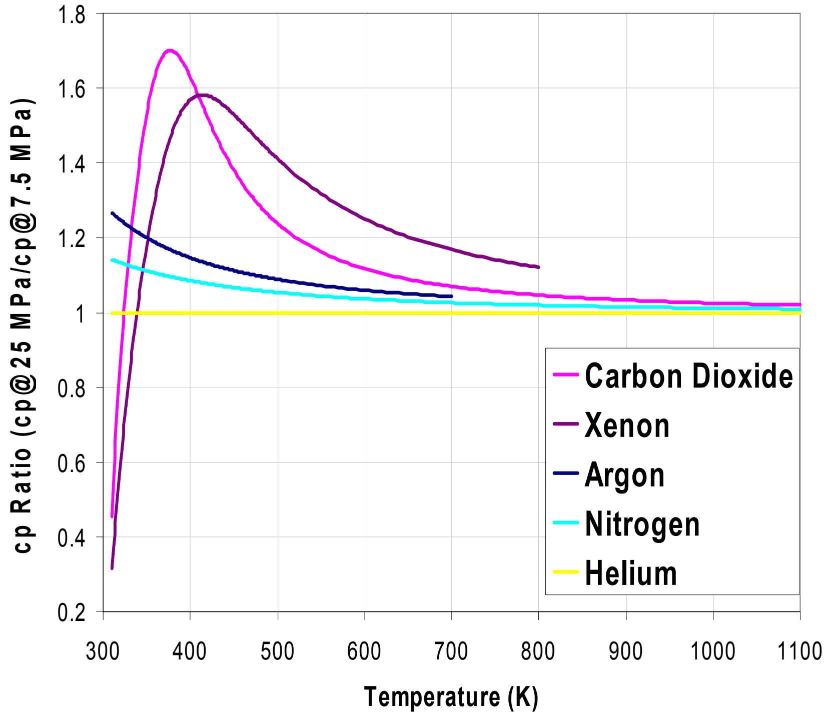
http://www.gepower.com/products/gas_turbines/7fb_intro.html

GE Turbines have steadily increased combined cycle efficiency



R. Anex, et al.,
http://www.ou.edu/spp/turbine/paper.html

AHTR Could Utilize Current Air Turbine Technology in a Closed N_2 Brayton Cycle as a Near Term Option

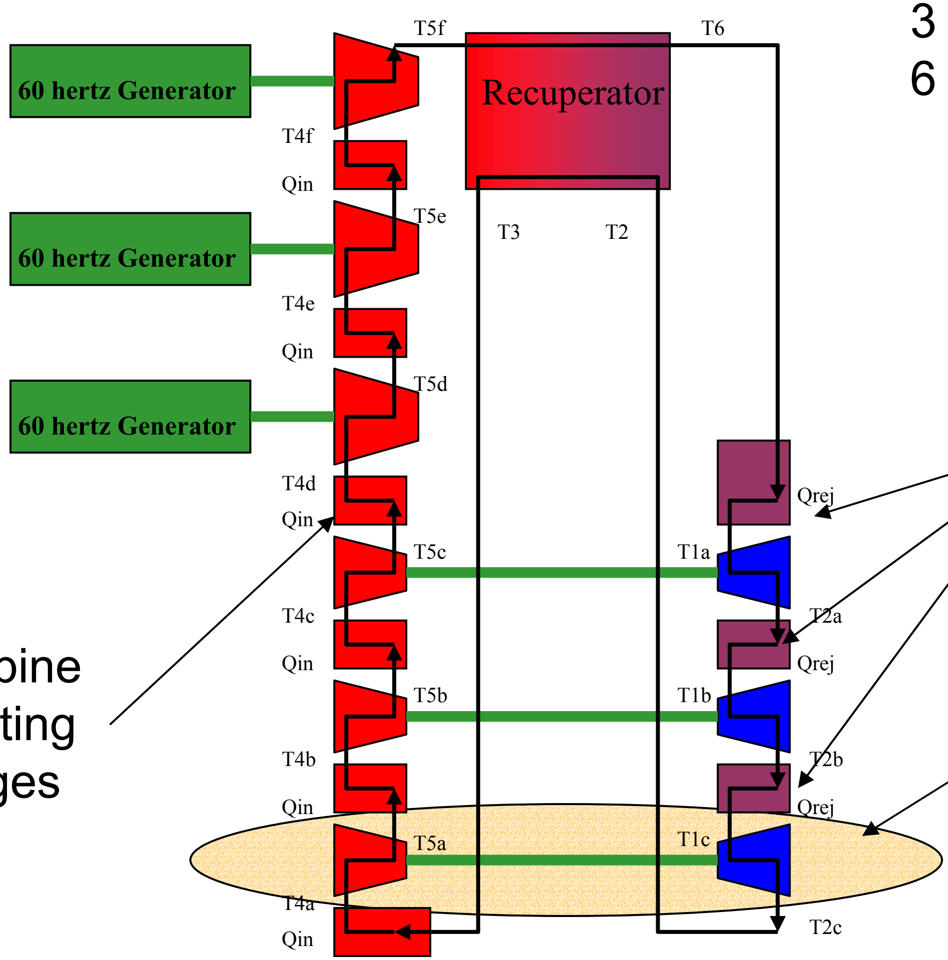


Constant Pressure Heat Capacity Ratio as a Funtion of Temperature for Various Brayton Cycle Working Fluids

- N_2 is nearly an Ideal Gas
- N_2 cycle has similar efficiency to He cycle
- N_2 turbine development could utilize air turbine technology
- Heat exchangers must be designed for lower N_2 heat transfer

Molten Salt Coolant (high heat capacity, low ΔT) Facilitates the Use of IC&H Technology and Independent Running Aircraft Turbine Technology

3 compressor stages,
6 turbine stages



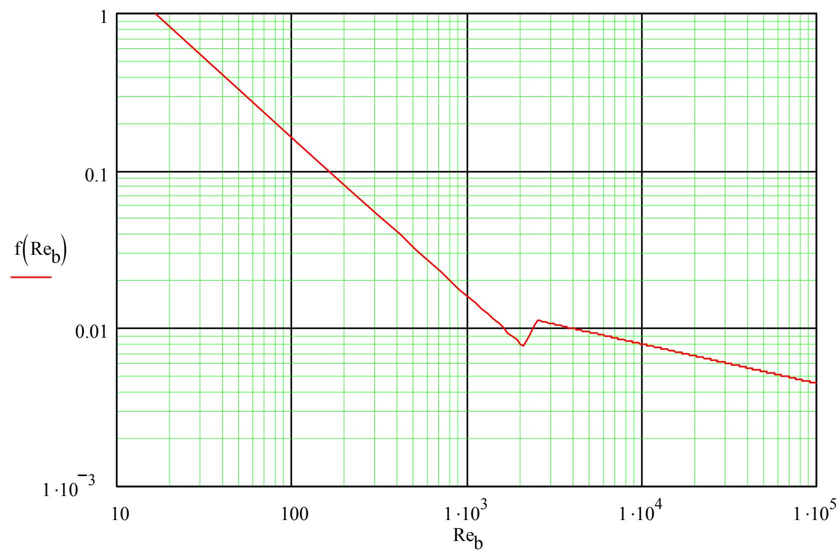
compressor cooling stages

Turbine heating stages

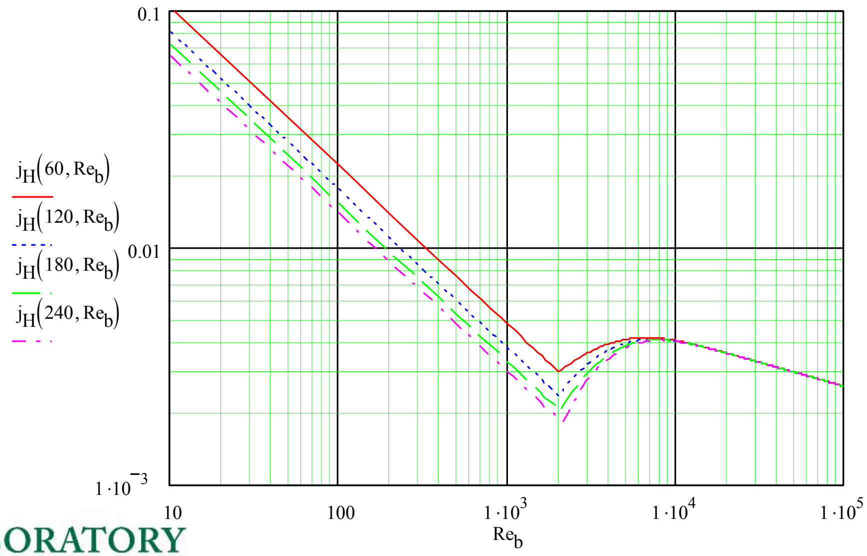
Turbine staging could be incorporated from aircraft technology, since many of the stages will not have to operate at 60 hertz

Design Parameter Used In Analysis

$turbine_{blade\_height}(min) = 5$ cm
$\Delta T_{across\ reactor} = 100K$
$\rho_{structural\ material} = 8\ g/cm^3$
$\sigma_{limit}(300K) = 68$ MPa
$\sigma_{limit}(1000K) = 20$ MPa
$\sigma_{limit}(1273K) = 10$ MPa
$eff_{turbine} = 93\%$
$eff_{compress} = 85\%$
$eff_{QinHX} = 98\%$
$eff_{QrejHX} = 98\%$
$eff_{QrecupHX} = 95\%$
$\Delta P/P_{recupHP} = 1\%$
$\Delta P/P_{recupLP} = 2\%$
$\Delta P/P_{Qin} = 1\%$
$\Delta P/P_{Qrej} = 1\%$

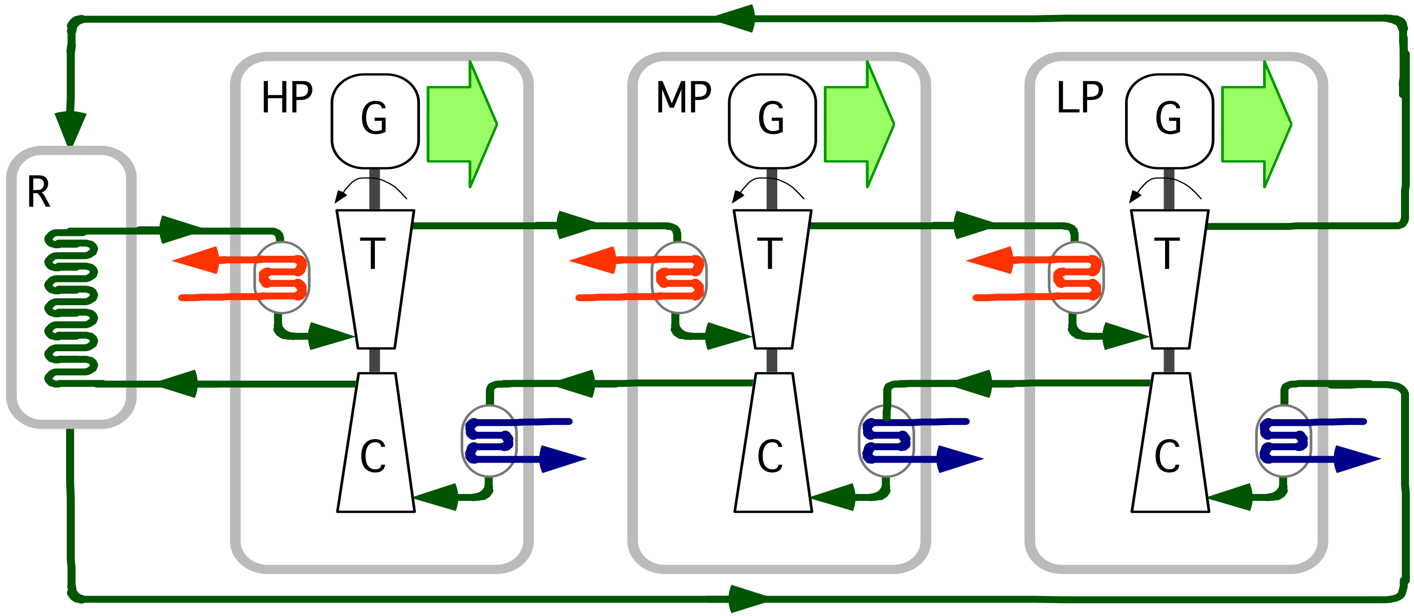


$$\frac{dP}{dx} := \frac{2 \cdot f \cdot mdot^2}{D_{hd} \cdot \rho \cdot A_{flow}^{\ 2}}$$

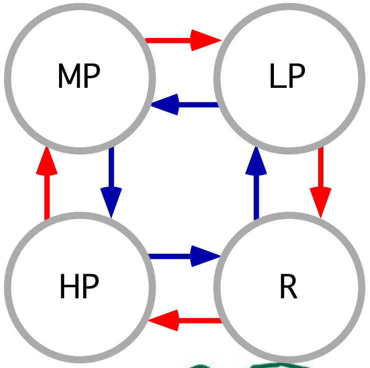


$$\frac{dT_g}{dx} := \frac{4 \cdot j_H \cdot \Delta T_{gw} \cdot \left(\frac{\mu_b}{\mu_w}\right)^{0.14}}{Pr_b^{\ .67} \cdot D_{hd}}$$

Layout for a 2400 MW(t) Helium Brayton Cycle Based on Three GT-MHR Power Conversion Units (PCUs)

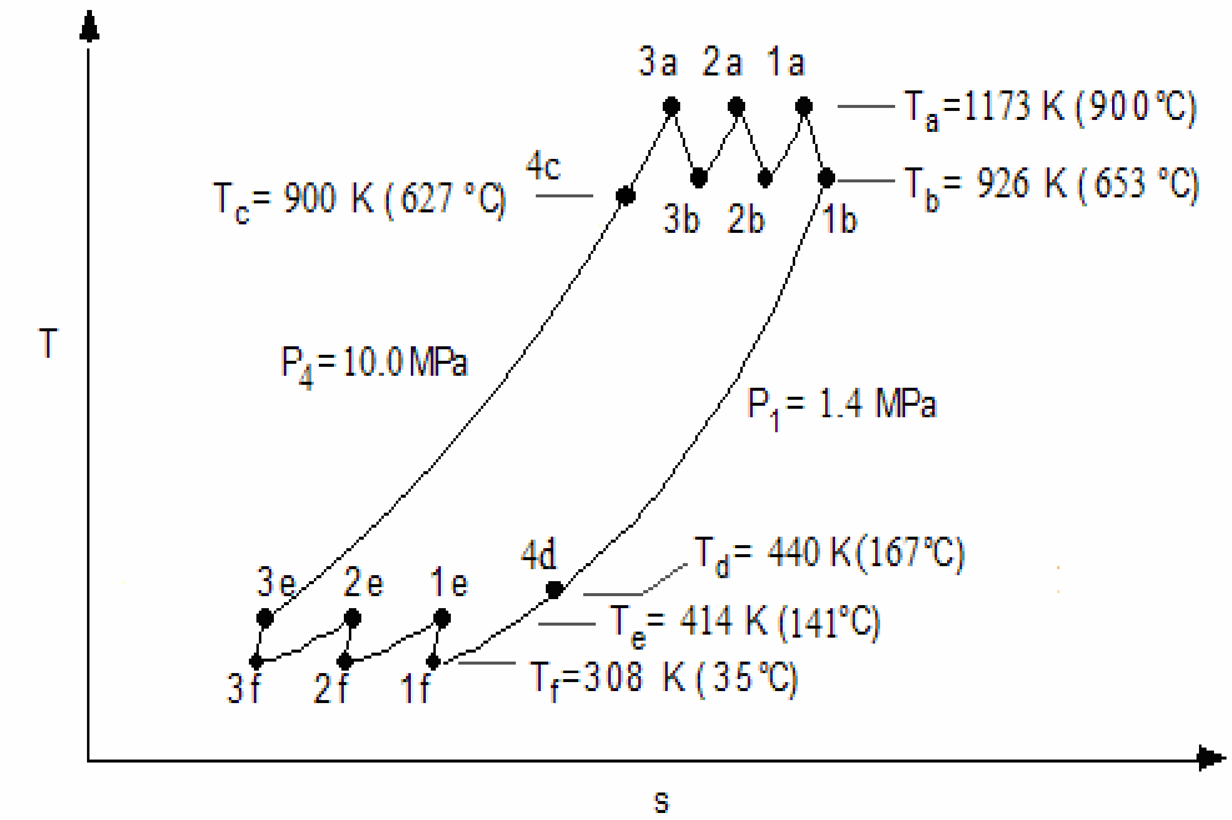


Schematic showing the MCGC, with n = 3 turbine (T) / compressor (C) / generator (G) modules, each with a cooler and heater, and a recuperator (R) located in a fourth vessel.

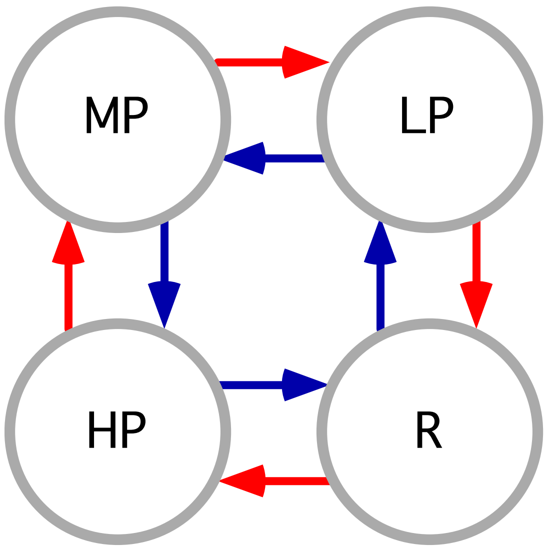


Schematic of physical arrangement of vessels and hot/cold leg flows

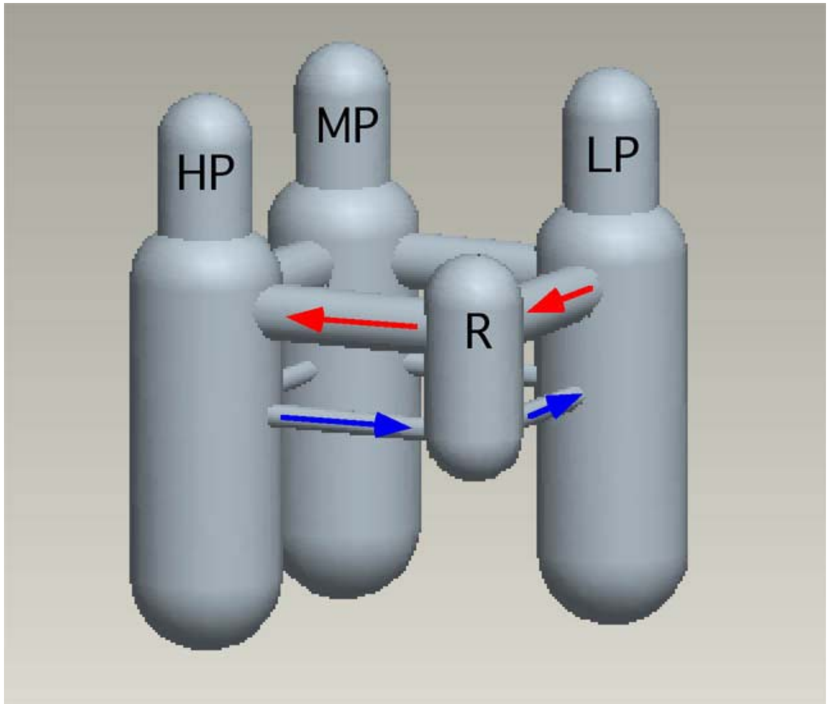
Temperature/Entropy Diagram for Helium Multi-reheat Brayton Cycle



Power Conversion Unit Arrangement Schematics



A very compact PCU arrangement is possible with a hot leg located above the cold leg. Hot leg hot duct operates at ~650°C, and ~10% of cold flow is bypassed upward to hot-leg annulus to cool the pressure boundary. ~90% of cold flow is transferred in lower cold leg, minimizing flow distance and pressure drop



Physical arrangement for the MCGC based on the GT-MHR PCU (vessels are ~ 30 m high)

The PCU Requires Only Modest Modifications To The Current GT-MHR PCU

Changes Required for MCGC

Generator becomes taller due to higher power output

Hot cross-over legs at turbine exit elevation

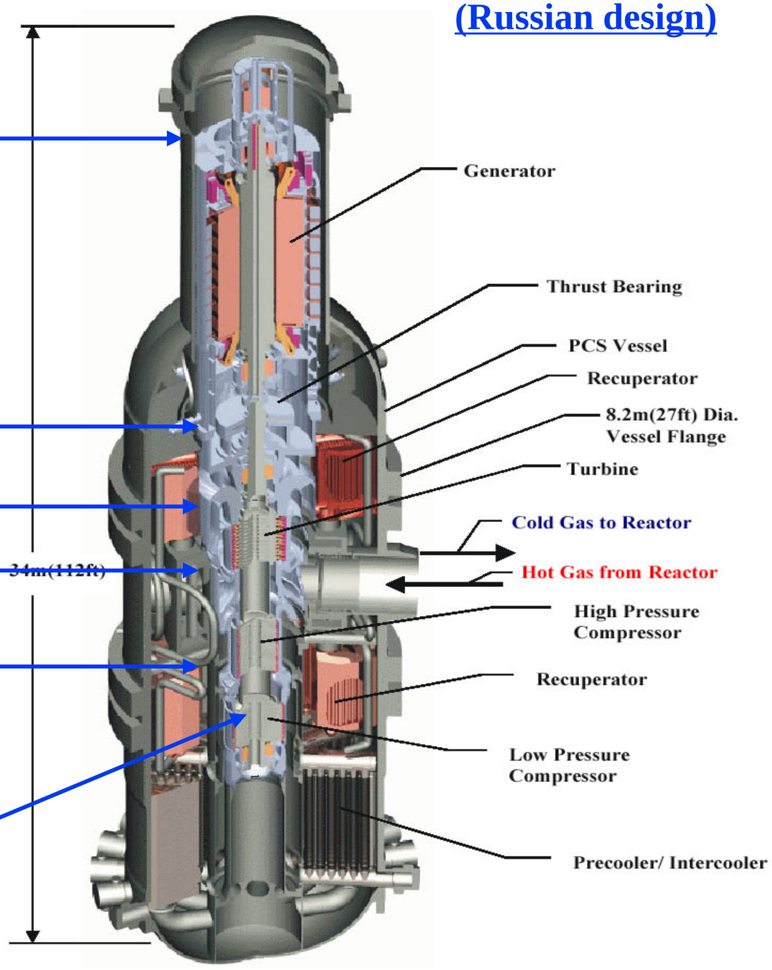
Heaters located in annular space around turbine

Cold cross-over legs at compressor exit elevation

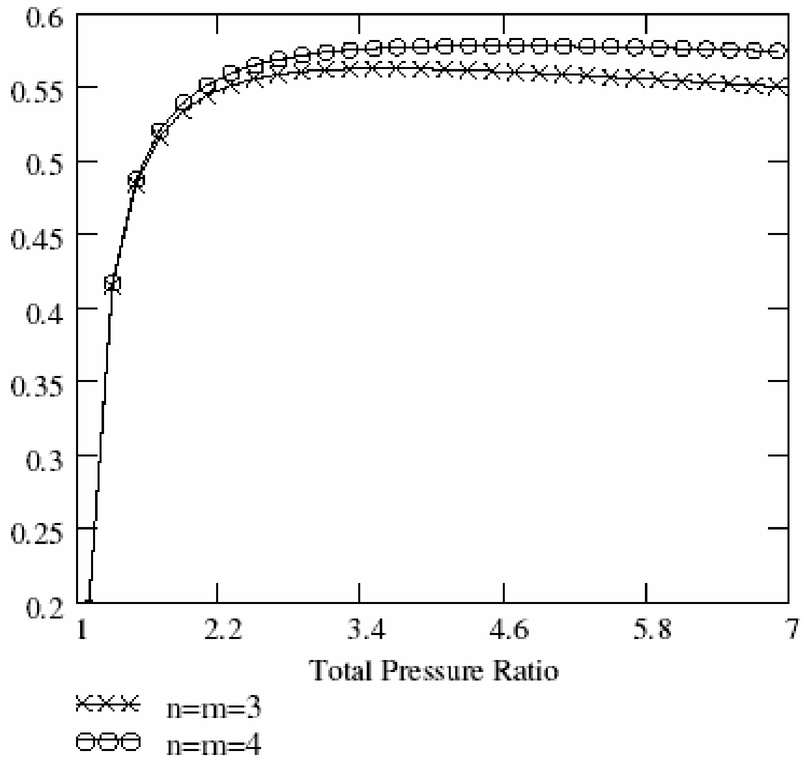
Single compressor/cooler (no intercooling), cooler moves up into annular space currently occupied by lower recuperator

With higher peak pressure (10 MPa vs. 7 MPa, the MCGC turbomachinery size is only slightly larger in diameter, and similar length, to GT-MHR

Current GT-MHR PCU (Russian design)



Thermal Efficiency Above 50% Is Achieved For a Wide Range of Pressure Ratios



Overall thermal efficiency versus total pressure ratio (for constant pressure loss P = 0.07)

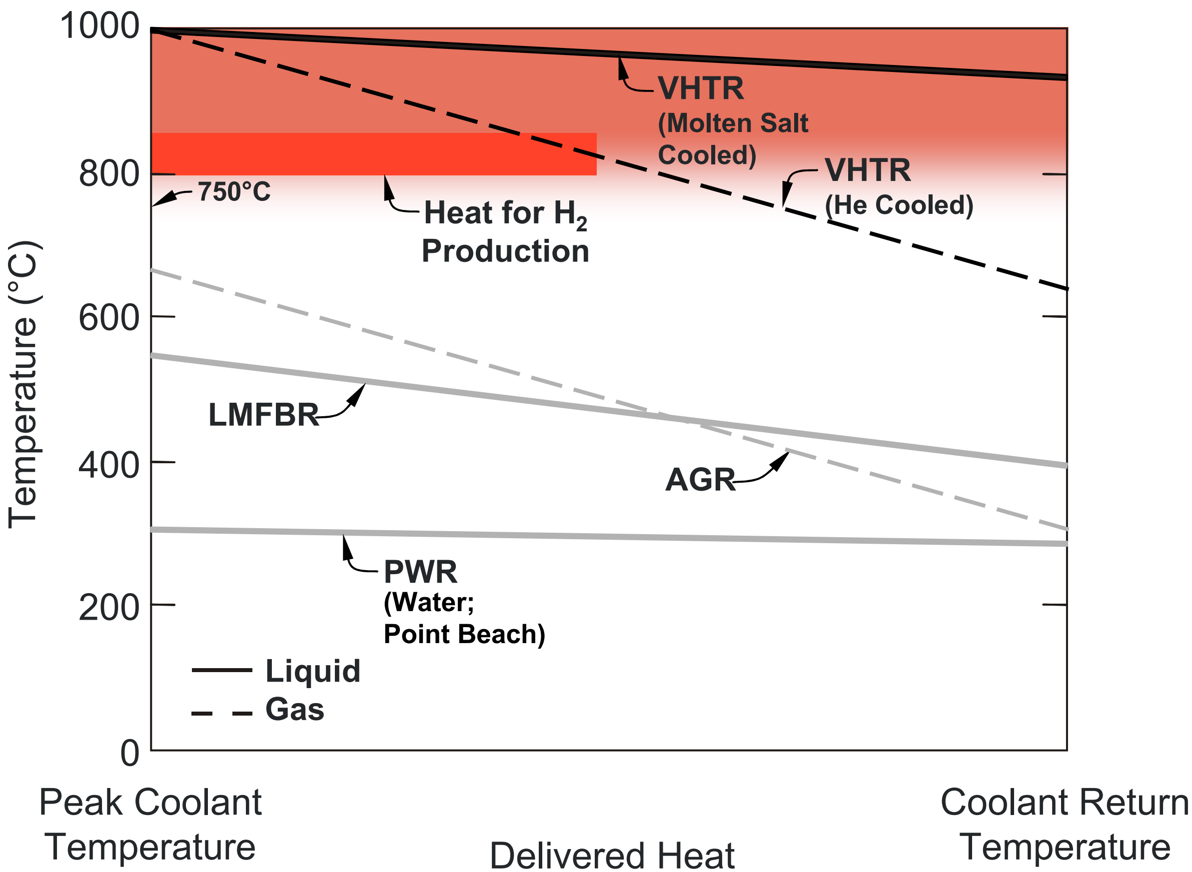
56% for optimal pressure ratio for 3 stages
Compare this value to:

- **Carnot cycle efficiency: 74%**
- **Current PWR & BWR: 34%**
- **GT-MHR: 48%**
- **Steam plant coal fired: 40%**
- **Gas turbine natural gas fired: 45-55%**

Thermal efficiency does not change significantly between pressure ratios of 3 and 7; higher pressure ratios give lower turbine exit and hot crossover leg temperatures, and more compact recuperators

Hydrogen Production Methods Require Much of the Heat To Be Delivered Over a Small Temperature Range

Liquid Cooling Can More Easily Achieve This Goal



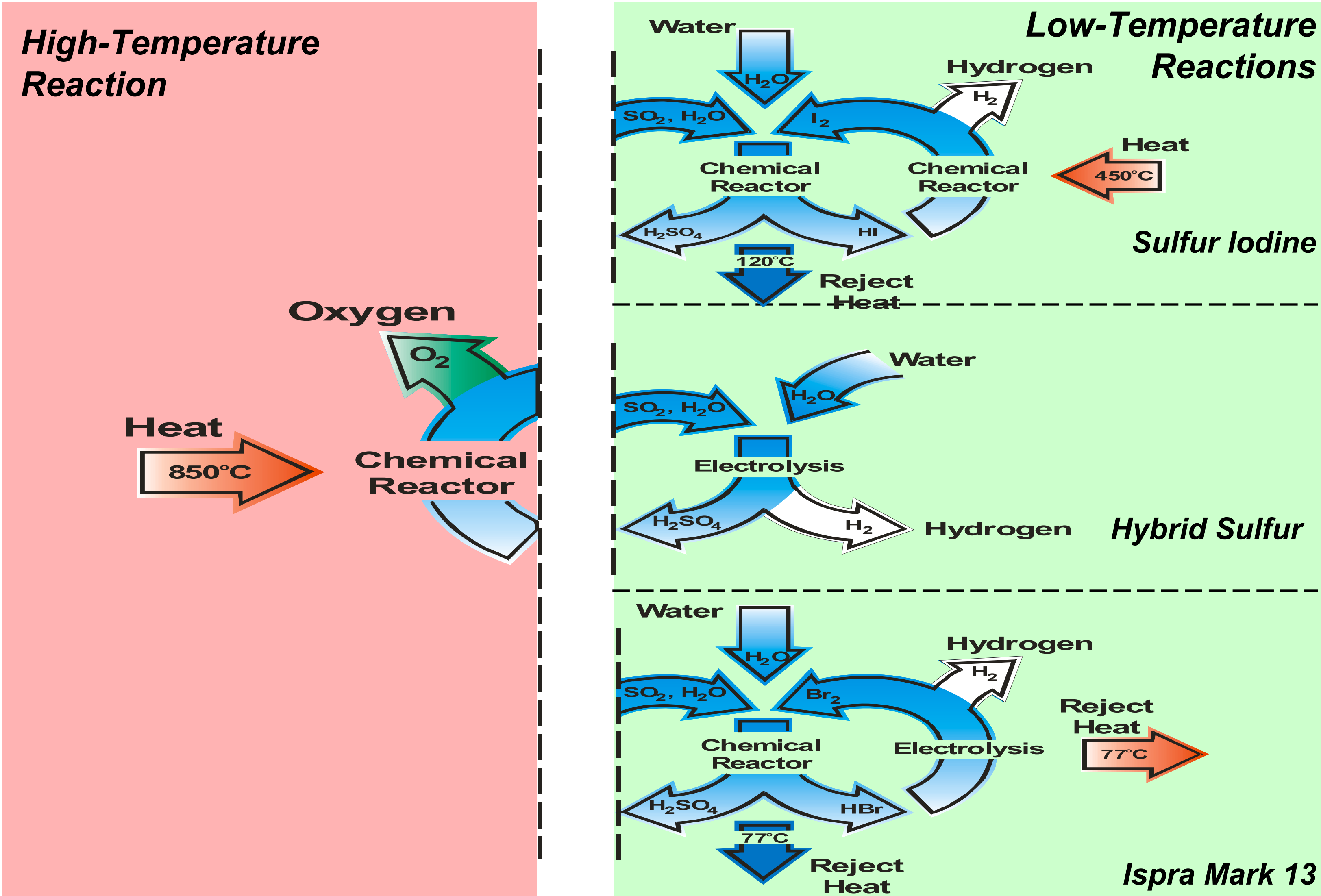
Hydrogen Production Methods

Leading candidates are thermochemical sulfur cycles

High-temperature step is the thermal decomposition of sulfuric acid

Requires most of the energy delivered at high temperatures

Sulfur Family of Thermochemical Hydrogen Production Cycles
(The Sulfur Cycles Include 3 Of The 4 Fully Demonstrated Cycles)



Economics Data

Initial AHTR Cost Analysis Is Based On Scaling and Combining Costs From S-PRISM and GT-MHR

	S-PRISM two blocks 1000 MWt per reactor indirect cyc 4 reactors	Block scaling exponents derived from ALMR 1-3 blocks	S-PRISM one block 2 reactors 1000 MWt per reactor	Estimated cost factors 2 to 1 reactors in same block	S-Prism one block one reactor 2000 MWt	Power scaling exponents 2000-2400 MWt	S-Prism one block one reactor 2400 MWt (912 MW_e) Indirect cycle	GT-MHR 910720/1 target costs 4x600 MWt (1145 MW_e) Direct cycle	**AHTR** **(Mix & Match)** **(S-Prism & GT-MHR** **at 2400 MWt)** (1145 MW_e) **Indirect cycle**	
	4000 MWt		2000 MWt		2000 MWt		2400 MWt	2400 MWt	**2400 MWt**	
	1996 k$		1996 k$				1996 k$	1994 k$	**1996 k$**	
Land	0		0				0	2,000	**0**	S Prism
Structures & Improvements	232,000	0.82	131,166	0.80	104,933	0.5	114,948	149,000	**114,948**	S Prism
Reactor Plant equipment	900,000	0.86	497,025	0.81	404,081	0.6	450,793	353,000	**450,793**	S Prism
Turbine Pant equipment	236,500	0.99	118,674	1.00	118,674	0.8	137,309	211,000	**154,743**	GT-MHT: 3 units
Electric Plant equipment	128,000	0.85	70,981	1.00	70,981	0.5	77,756	65,000	**67,706**	GT-MHT
Misc plant equipment	39,000	0.51	27,464	0.80	21,971	0.3	23,206	31,000	**32,291**	GT-MHT
Main heat reject system	38,500	0.88	20,976	1.00	20,976	0.8	24,270	35,000	**28,479**	GT-MHT
Special Materials	20,000	1.0	10,000	1.00	10,000	0.8	11,570		**11,570**	S Prism
total direct cost	1,594,000		876,286		751,616		839,853	846,000	**860,530**	
Construction Services	138,000	0.71	84,365		72,362	0.4	77,837	107,000	**79,753**	S-Prism %
Home Office Engineering & S	69,000	0.34	54,383		46,646	0.2	48,378	68,000	**49,569**	S-Prism %
Field Office Engineering & S	79,000	0.71	48,402		41,516	0.4	44,657	52,000	**45,756**	S-Prism %
Owner's Cost	290,000	0.83	163,017		139,824	0.4	150,402	150,000	**154,105**	S-Prism %
total indirect	576,000		350,167		300,348		321,274	377,000	**329,184**	
Base Construction Cost	2,170,000		1,226,452		1,051,964		1,161,127	1,223,000	**1,189,714**	
Contingency	0		0		0		0	295,000	**0**	S-Prism %
Overnight Cost	2,170,000		1,226,452		1,051,964		1,161,127	1,518,000	**1,189,714**	
electrical power, MWe	1520		760		760		912	1145	**1145**	GT-MHR
$/kWe, Overnight	1,428		1,614		1,384		1,273	1,326	**1,039**	
2002_$/kWe, Overnight	1,580		1,786		1,532		1,409	1,528	**1,150**	
Relative_$/kWe, Overnight	1.00		1.13		0.97		0.89	0.97	**0.73**	