Molten Salt Reactors:

A New Beginning for an Old Idea

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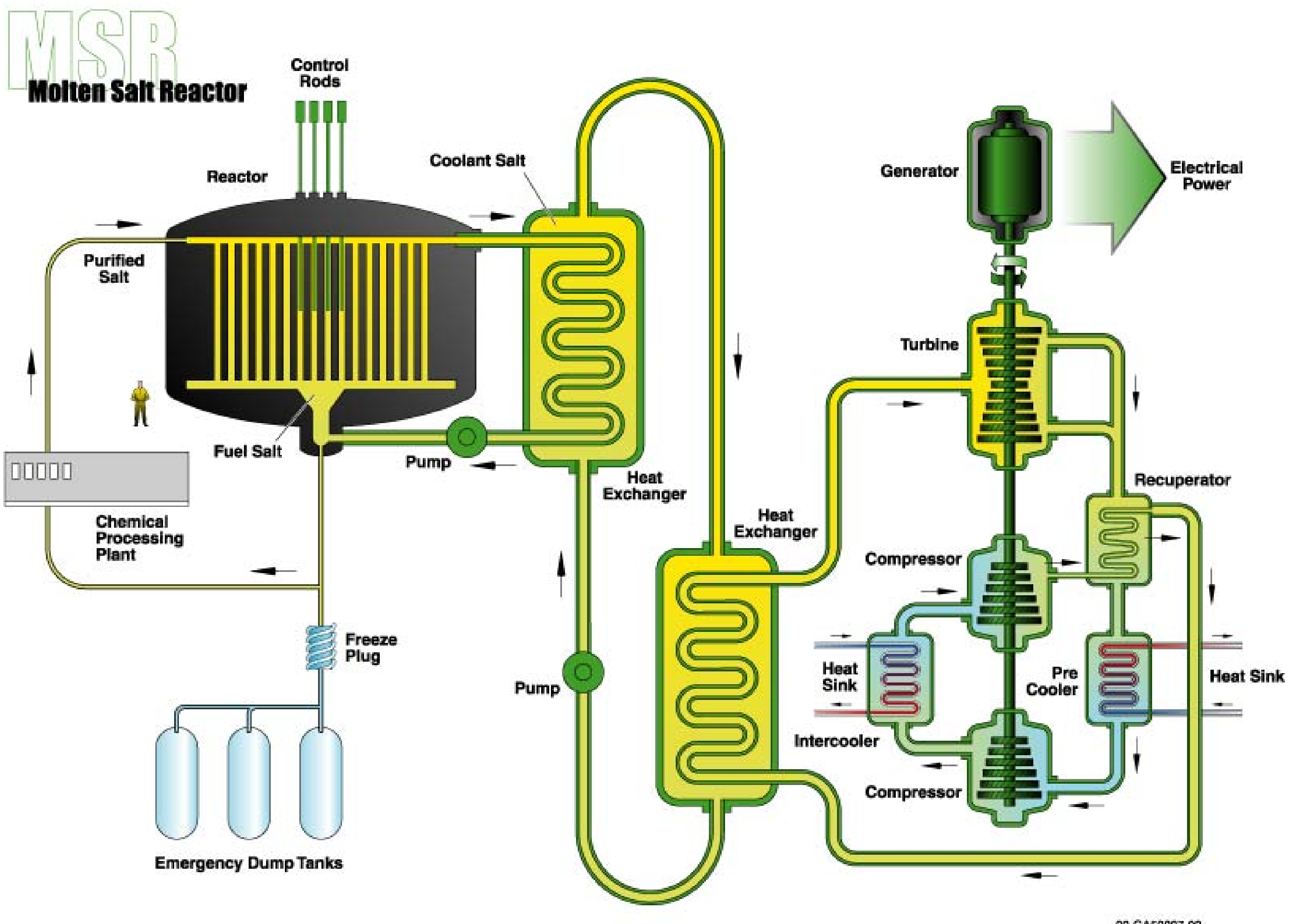
Chalk River Nuclear Laboratories

David LeBlanc

Physics Department, Carleton University
& Ottawa Valley Research Associates Ltd.

The Single Fluid, Graphite Moderated Molten Salt Breeder Reactor (MSBR)



General Attributes of the "Traditional" MSBR Design

- $2\,^{7}LiF\text{-}BeF_2$ - 12% ThF_4 - 0.3% $^{233}UF_4$
- 700 ^{o}C Outlet Temperature
- 44.4% on Steam Rankine cycle
- Up to 48% on Gas Brayton cycle
- Fission Products removed on 20 day cycle
- ^{233}Pa removed on 10 day cycle
- Specific Inventory of 1500 kg
- Breeding Ratio of 1.06
- 20 year doubling time

General Benefits of Any Molten Salt Design

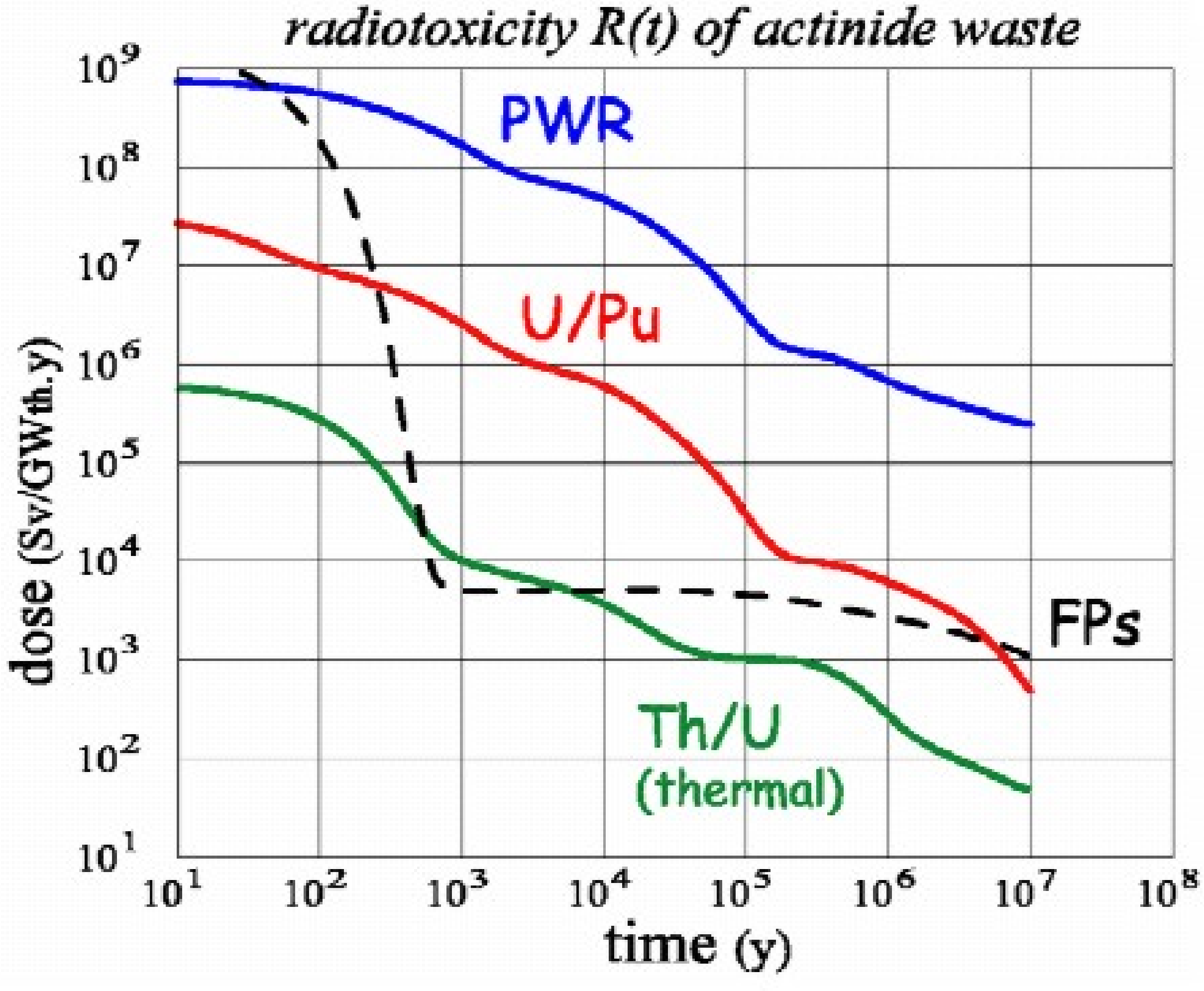
- Salts have high boiling point and operate at low pressure
- Fuel salt at the lowest pressure of the circuit, the opposite of a LWR
- Control rods or burnable poisons not required so very little excess reactivity
- Volatile fission products continuously removed and stored, including Xenon.
- Freeze plug melts upon fuel overheating to drain to critically safe, passively cooled dump tanks

General Benefits of Any Molten Salt Design

- Low Fissile Inventory
- Very high thermal efficiency
- Ideal for LWR TRU waste destruction
- Ability to use closed thorium cycle
  - Only consume 800 kg thorium per GWe year
  - Transuranic waste production extremely low
  - Much lower long term radiotoxicity

Radiotoxicity PWR vs FBR* vs MSR*

*Assuming 0.1% Loss During Processing



Problems Specific to the Single Fluid, Graphite Moderated MSBR

- Limited graphite lifetime (4 years)
- Fuel processing hindered by chemical similarity of thorium and rare earth fission products
- Problem with temperature reactivity coefficient recently discovered
- ^{233}Pa separation adds a unique proliferation concern

Proliferation Basics

- ^{232}U present in significant quantities
  - 64 year half-life with 2.6 MeV gamma ray
  - Makes illicit use or diversion extremely difficult and highly detectable
- ^{233}Pa removal can lead to “clean” ^{233}U and thus should be avoided
- If HEU use proves a “non-starter”, adding depleted uranium can render the cycle denatured and thus LEU (at a cost…)
- Any Plutonium present is extremely poor quality (much ^{238}Pu) and hard to extract

2 Fluid Versus Single Fluid

- A 2 Fluid design has separate fluids
  - A Fuel Salt containing $^{233}UF_4$
  - A Blanket Salt containing ThF_4
- Advantages;
  - Absence of thorium in fuel salt makes fission product removal far easier
  - ^{233}Pa removal unnecessary
  - Strongly negative temperature and void coefficients for the fuel salt

And just to confuse you
The 1 and ½ Fluid Design

- A mixed salt of $^{233}UF_4$ and ThF_4 in the core
- Surrounded by a ThF_4 blanket salt to prevent neutron leakage
- *Has some of the advantages of both designs but many disadvantages of both as well*

Choice of Neutron Spectrum

- Harder Spectrum
  - Losses to fission products drop
  - Prompt neutron lifetime is shorter
  - Typically requires higher fissile load
  - Lower cross section of Pu means the limited solubility of PuF_3 an issue
- Softer Spectrum
  - Fissile inventory can be very low
  - Losses to fission products highest
  - Any extra structural material must be very limited or breeding not possible
  - Prompt neutron lifetime is increased

Fuel Processing Basics

- Fluoride Volatility Process
  - Removes uranium from carrier salt
- Fission Product Removal
  - Essential if breeding desired
  - Several year cycle possible in some reactor designs
- Protactinium Removal and Hold Up
  - Non essential but improves breeding

Fluoride Volatility

- Simple bubbling of fluorine gas through carrier salt
- Converts UF_4 to gaseous UF_6
- Conversion back to UF_4 done by adding H_2
- Well established process, known since 1950s

Fission Product Removal

- 1950 to 1964
  - Several potential methods but none proven to be practical
- 1964 Vacuum Distillation
  - Simple system to evaporate off carrier salt and leave behind fission products
  - Thorium would remain with fission products, so seen as only for a 2 fluid design

Fission Product Removal

- 1968 Liquid Bismuth Reduction
  - Ability to process for fission products with thorium present
  - Thorium behaves much like rare earth fission products which makes the process “delicate”
  - Complex and costly but works for Single Fluid design

The "Pa" Problem

- Protactinium Removal and Hold Up
  - Same liquid bismuth reductive reaction method used
  - Must be rapid to have effect on 27 day half life Pa, 3 to 10 days typical
  - Pa sent to holding tanks to await decay into ^{233}U
  - Adds unique proliferation concerns

Fuel Processing Summary

- For 2 Fluid design, choice of:
  - Vacuum Distillation
  - Salt Discard
  - Simplified Liquid Bismuth Extraction
  - Can skip Pa removal by increasing blanket volume
- For Single Fluid Design
  - Multi-staged Liquid Bismuth Extraction
  - Pa removal very difficult to avoid

A Strange Beginning
An Aircraft Reactor?

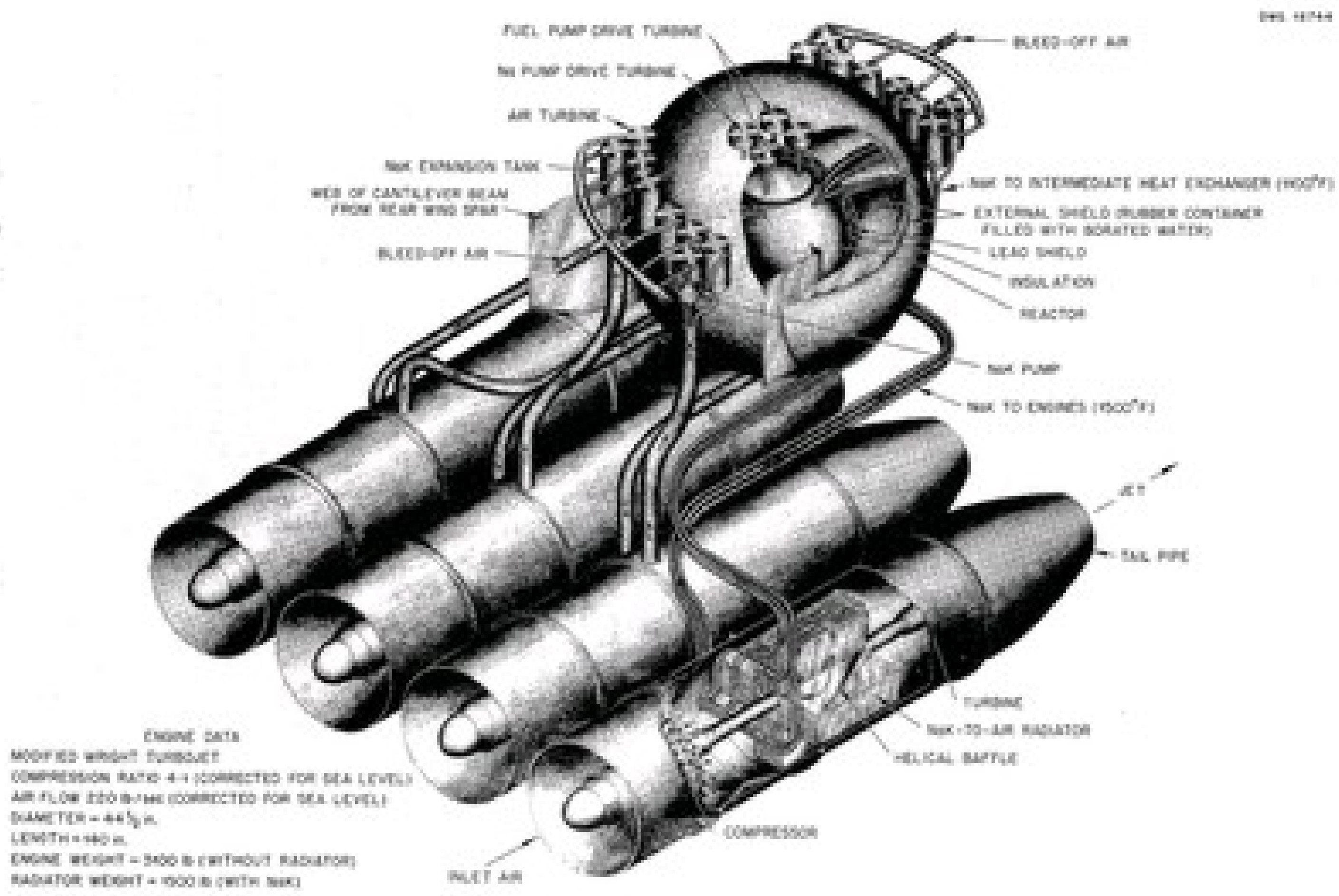


Fig. 4.33. Aircraft Power Plant (200 Megawatt).

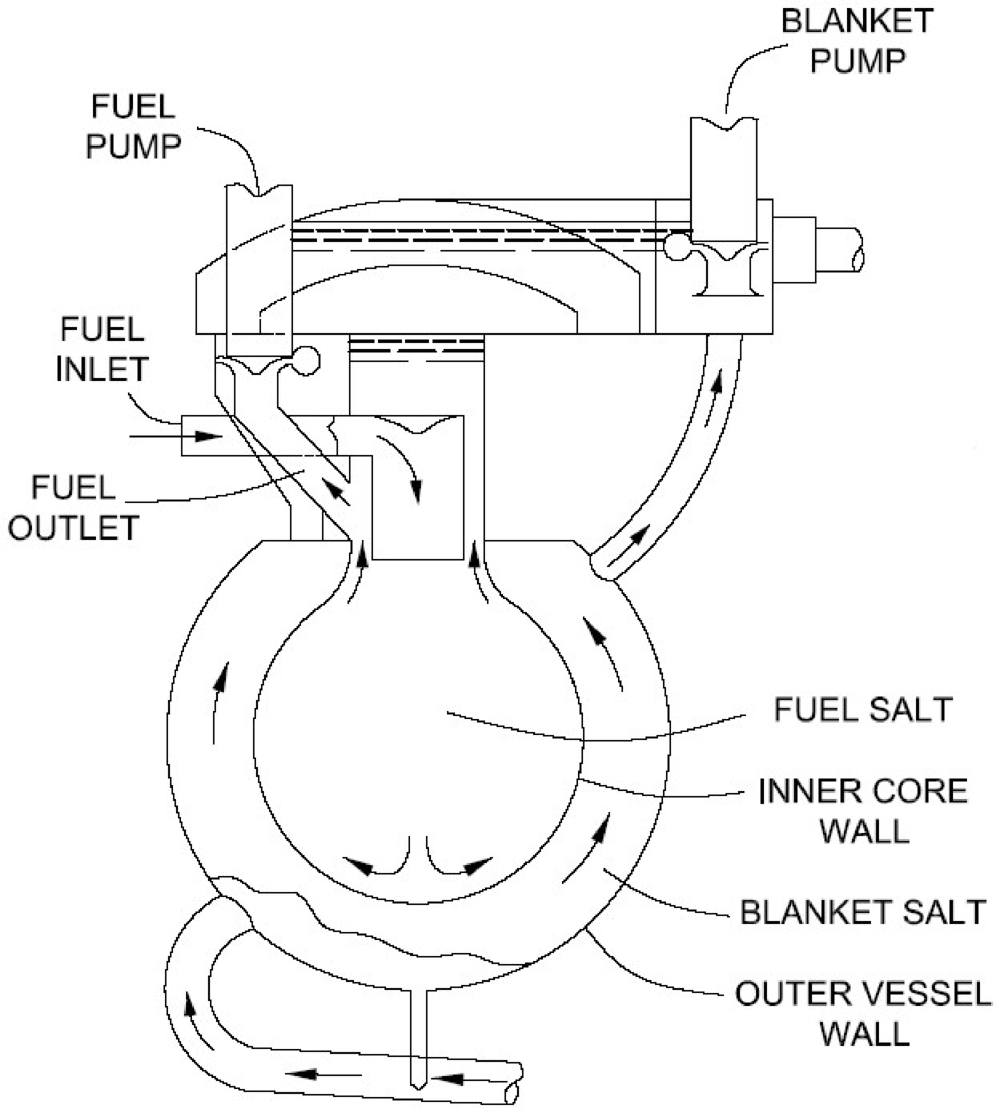
The Aircraft Reactor Experiment

- Test reactor of early to mid 1950s
- Up to 3 MWth during operation
- Very high temperature 860 ^{o}C
- Canned BeO moderator
- $NaF\text{-}ZrF_4$ carrier salt
- Points the way to possible power reactors

The Rise and Fall of the 2 Fluid Reactor

- Late 50s
  - Main focus on homogeneous reactors (no graphite or other solid moderator)
  - Looked at both ^{235}U converter reactors and thorium breeders
  - Carrier Salt itself provides significant neutron moderation
  - All studies are spherical geometry with 1/3 inch Hastelloy N core walls

Homogenous Molten Salt Reactor
Late 50s ORNL



Two Region Homogeneous Reactor

Projected breeding ratios assume thicker blanket and alternate barrier. From ORNL 2551, 1958

Core Diameter	3 feet	4 feet	4 feet	8 feet
ThF_4 in fuel salt mole %	0	0	0.25	7
^{233}U in fuel salt mole %	0.592%	0.158%	0.233%	0.603%
Salt Losses	0.087	0.129	0.106	0.087
Core Vessel	0.090	0.140	0.109	0.025
Leakage	0.048	0.031	0.031	0.009
Neutron Yield	2.193	2.185	2.175	2.20
Breeding ratio (Clean Core)	0.972	0.856	0.929	1.078
Projected B.R. (thinner wall)	*1.055*	*0.977*	*1.004*	*1.091*
Projected B.R. (carbon wall)	*1.105*	*1.054*	*1.066*	*1.112*

The Rise and Fall of the 2 Fluid Reactor Early 1960s

- Graphite now proven to be compatibility with fluoride salts
- Lower possible fissile inventory leads ORNL to change to a graphite moderated 2 fluid design
- Simple Sphere-Within-Sphere design would only allow about a 1 meter core
  - Obviously too low a total power output
- Complex fluid intermixing deemed necessary

1960`s, 2 Fluid, Graphite Moderated MSBR

- Intermixing of Fuel and Blanket salt in core done by graphite “plumbing”
- Allows large core diameter for adequate power production
- Graphite first shrinks and then swells under neutron irradiation
- Extremely difficult “Plumbing Problems” to solve

UNCLASSIFIED
ORNL-DWG 64-6959R

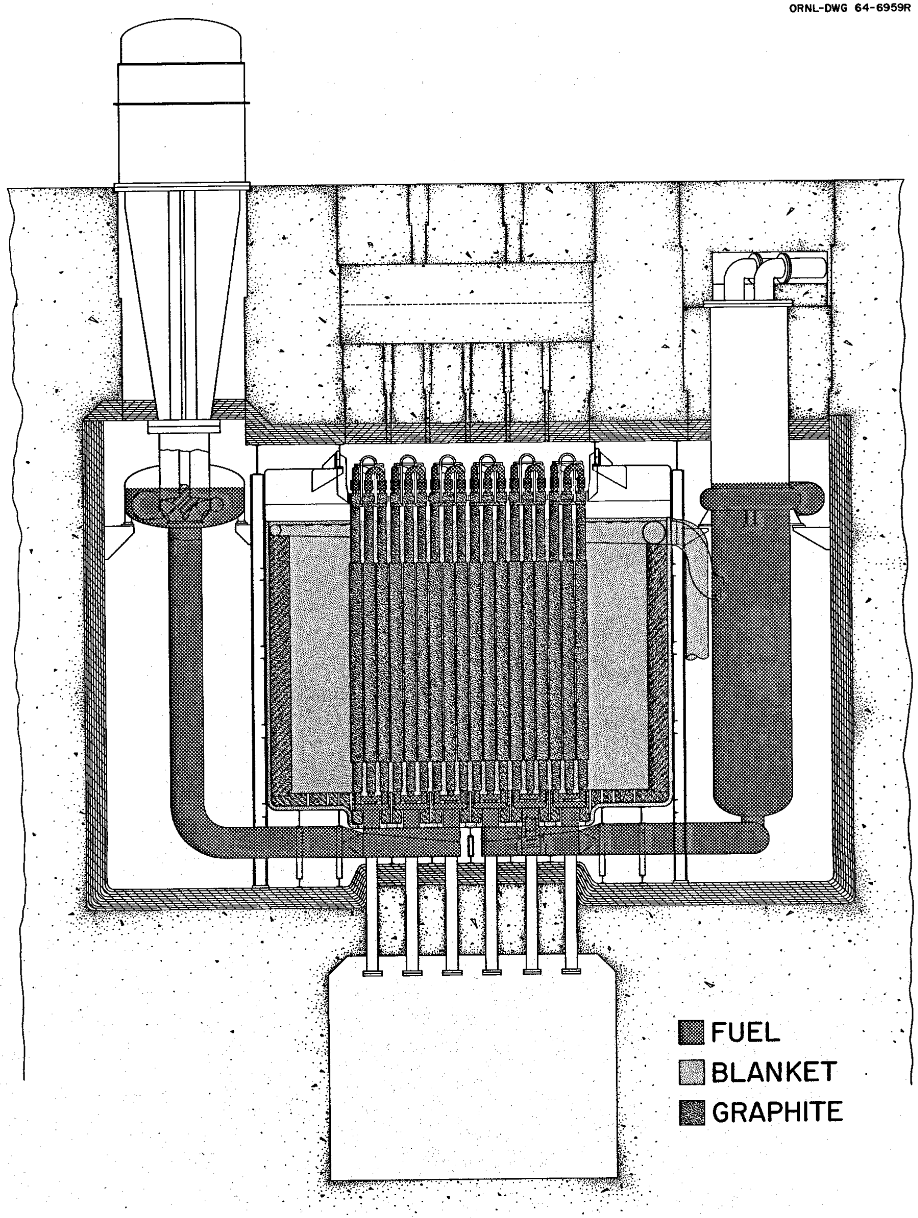
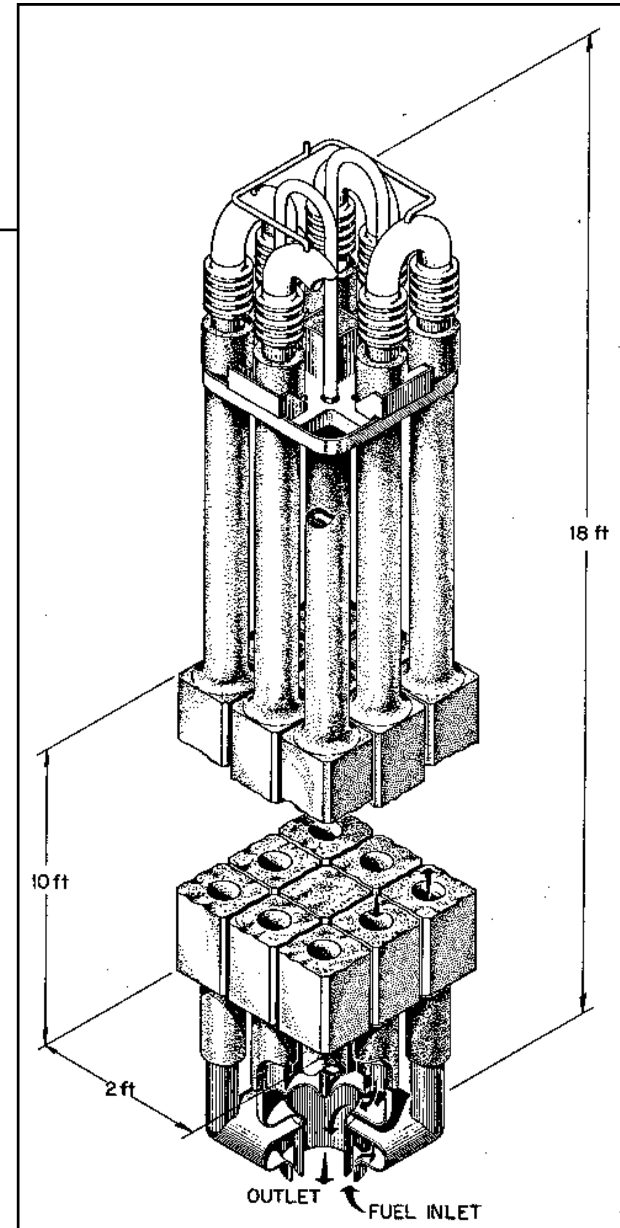


Fig. 2. Two-Region Molten-Salt Breeder.



Meanwhile, also in the mid 60s… Molten Salt Reactor Experiment MSRE

MSRE 8 MW(th) Reactor

- Chosen to be Single Fluid for simplicity
- Graphite moderated, 650 ^{o}C operation
- Designed from 1960 to 1964
- Start up in 1965
- Ran very successfully for 5 years
- Operated separately on all 3 fissile fuels, ^{233}U , ^{235}U and Pu
- Some issues with Hastelloy N discovered and solved

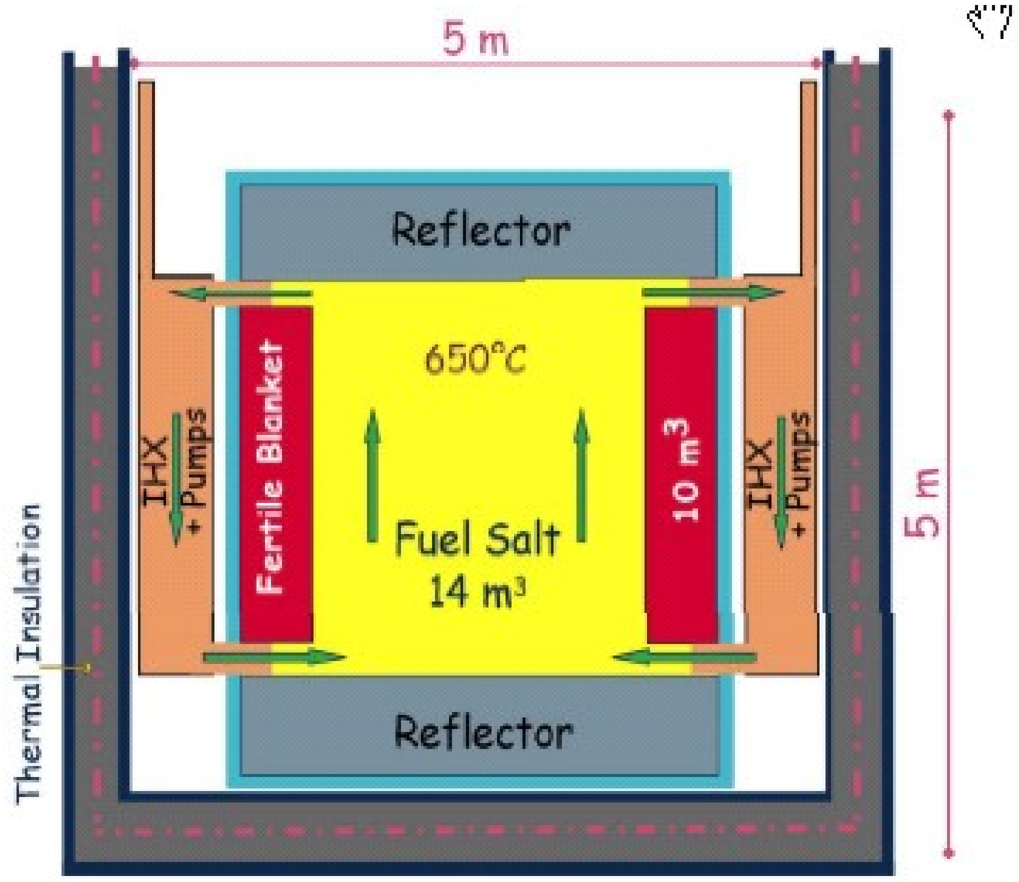
1968: The Start of the Single Fluid Era

- New fuel processing method that while far more difficult, can work with thorium in the fuel salt
- 2 Fluid concept abandoned
- “Plumbing Problem” left unsolved
- Major funding for MSBR cancelled in early 70s
- 2 Fluid concept largely forgotten

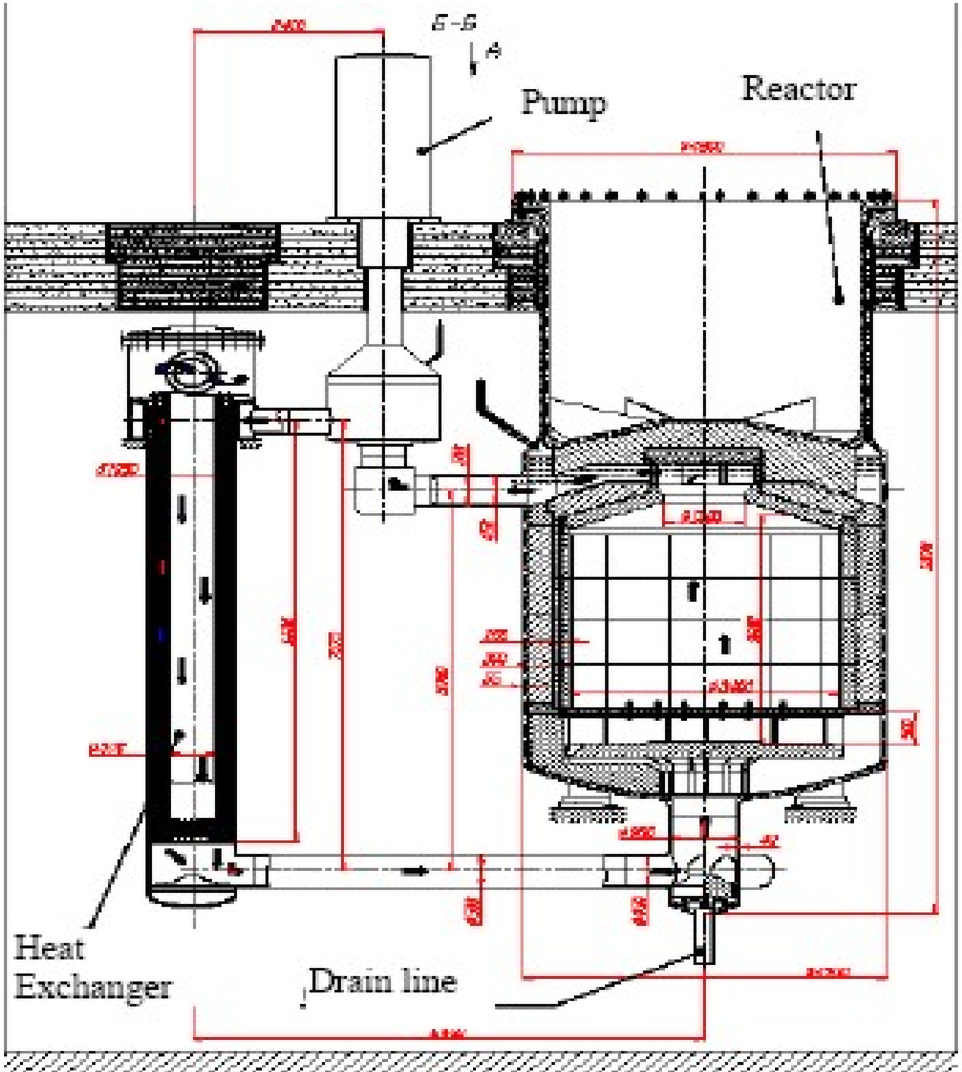
From the 70s to Today

- Most Recently
  - Major re-examination by large group in France
    - Find reactivity problem with 70s MSBR
    - Look to decrease processing needs by allowing breeding ratio to drop
    - Calculations show significant advantages of a harder spectrum without graphite
  - U.S. based effort to examine the use of clean salts as coolant for TRISO type fuels. Many advantages if positive void can be avoided
  - Much worldwide work on transuranic waste burning

The French TMSR
Thorium Molten Salt Reactor



Russian MOlten Salt Actinide Recycler and Transmuter MOSART



A Re-examination of MSBR Evolution

- Original development heavily focused on short doubling times to compete with LMFBR
  - Perhaps addition of graphite was a wrong choice?
- Is there a modern solution to the “plumbing problem”
- **OR** perhaps there never was one???

Restating the 2 fluid Plumbing Problem

- Sphere-Within-Sphere is simplest approach (or short cylinders)
- BUT, if inner fuel salt lacks thorium then the critical diameter must remain small (approx 1 m with or without graphite)
- Too low power producing volume
- “Standard” conclusion is fuel and blanket salts must be interlaced

What is the solution?
Here's a hint…

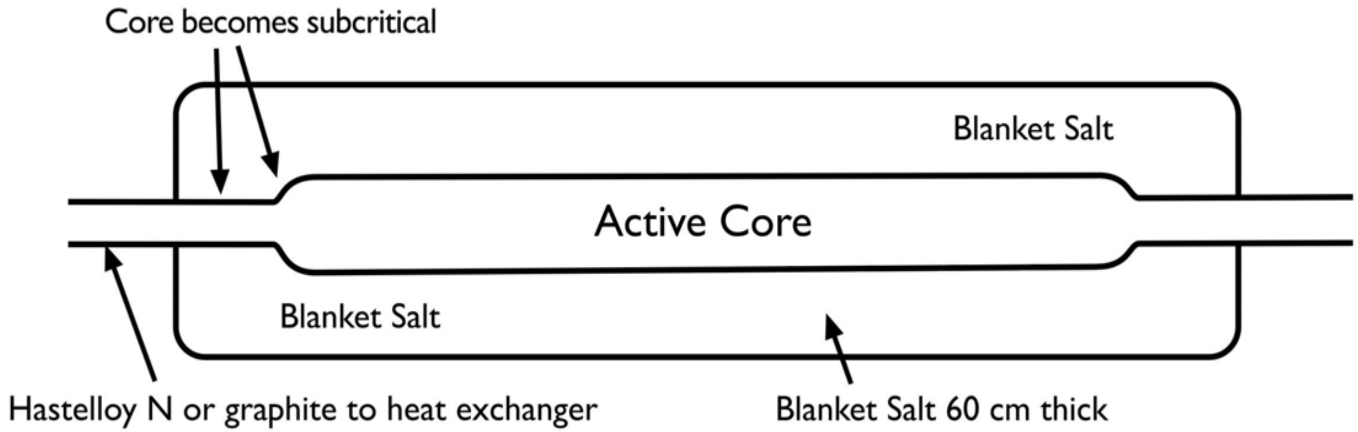
Geometry	Buckling^2	Ratio to B_{sphere}
Sphere	$\left(\frac{\pi}{R}\right)^2$	1
Infinite Cylinder	$\left(\frac{2.405}{R}\right)^2$	0.766
Finite Cylinder	$\left(\frac{2.405}{R}\right)^2 + \left(\frac{\pi}{H}\right)^2$	If $H = 10\,R$ 0.772
Infinite Slab	$\left(\frac{\pi}{a}\right)^2$	0.5

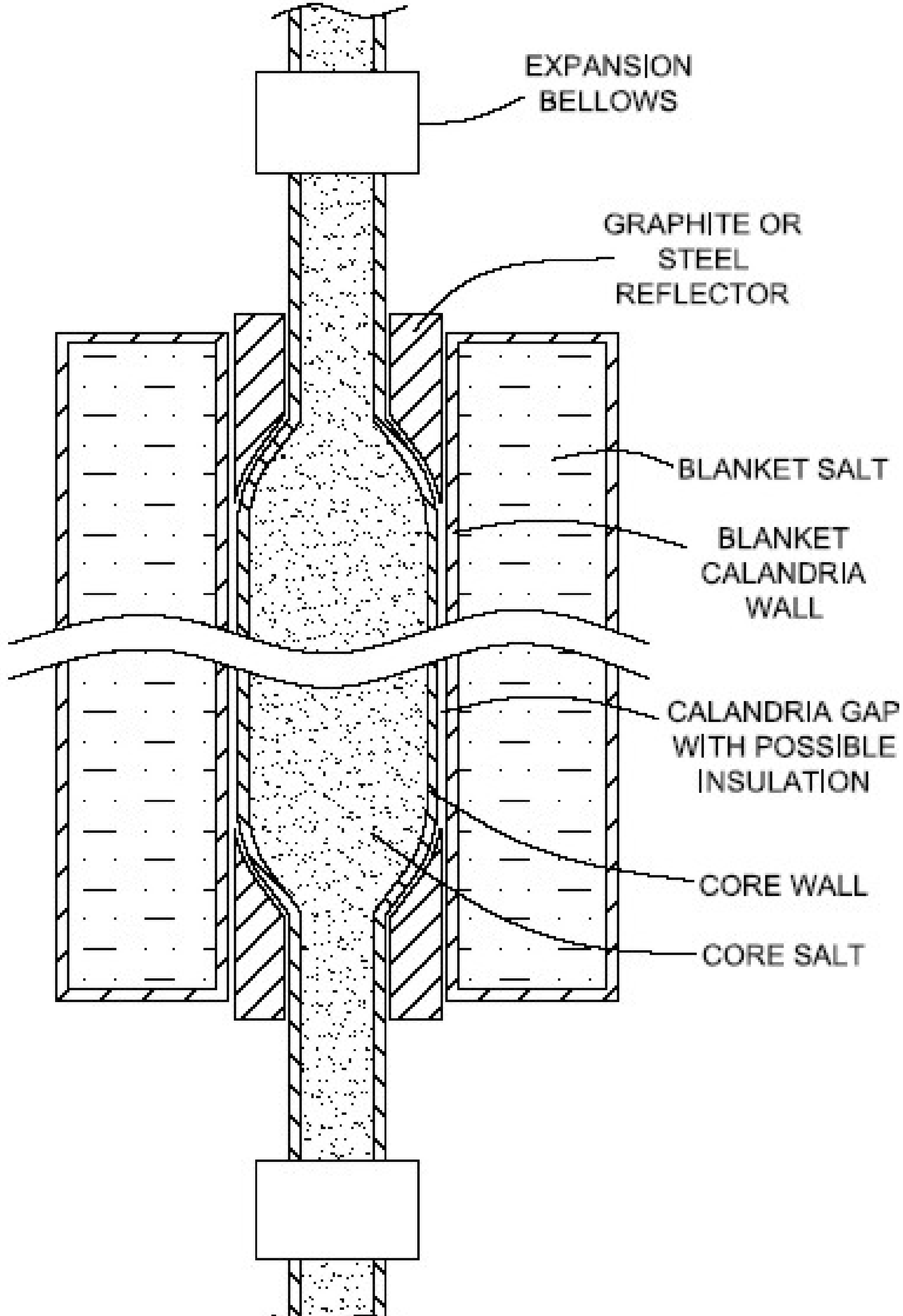
Modified Geometry 2 Fluid Reactor*
"Tube-Within-Shell"
*Patent Pending

Side View of Reactor Core and Surrounding Blanket Salt

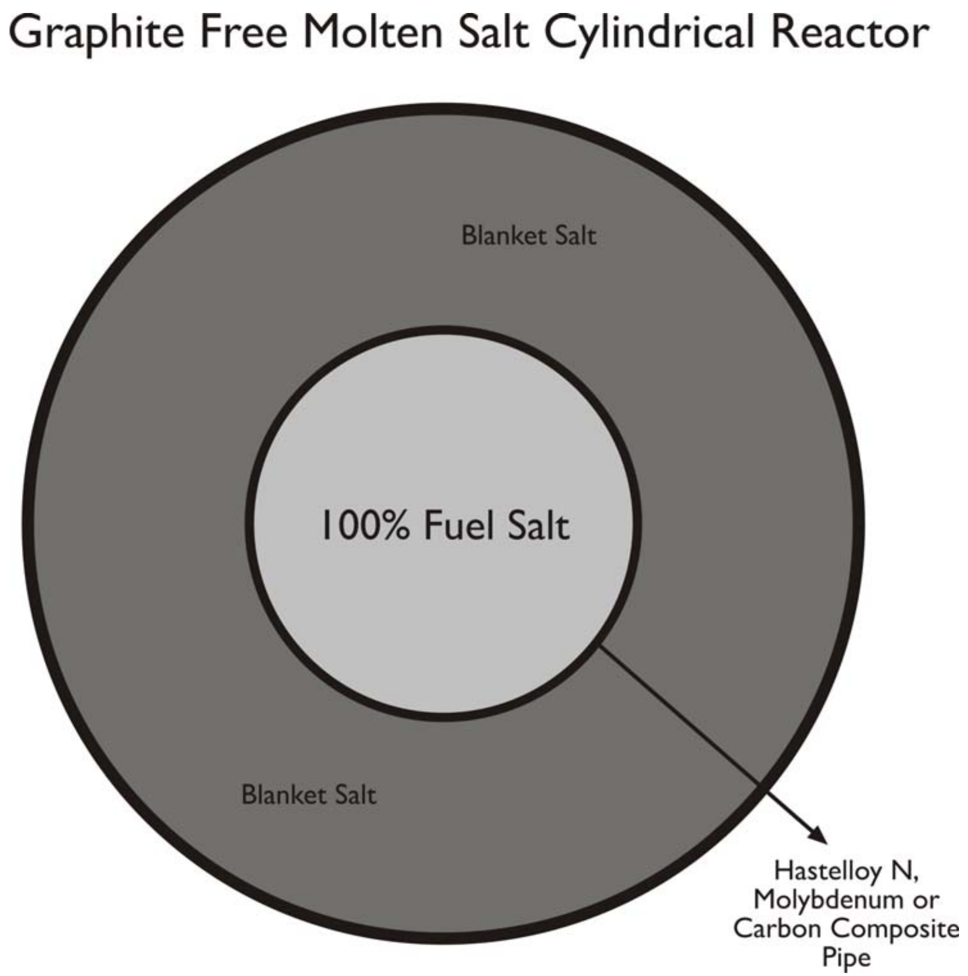
Core is Graphite + Fuel Salt or 100% Fuel Salt

Typical Diameter of 1 meter

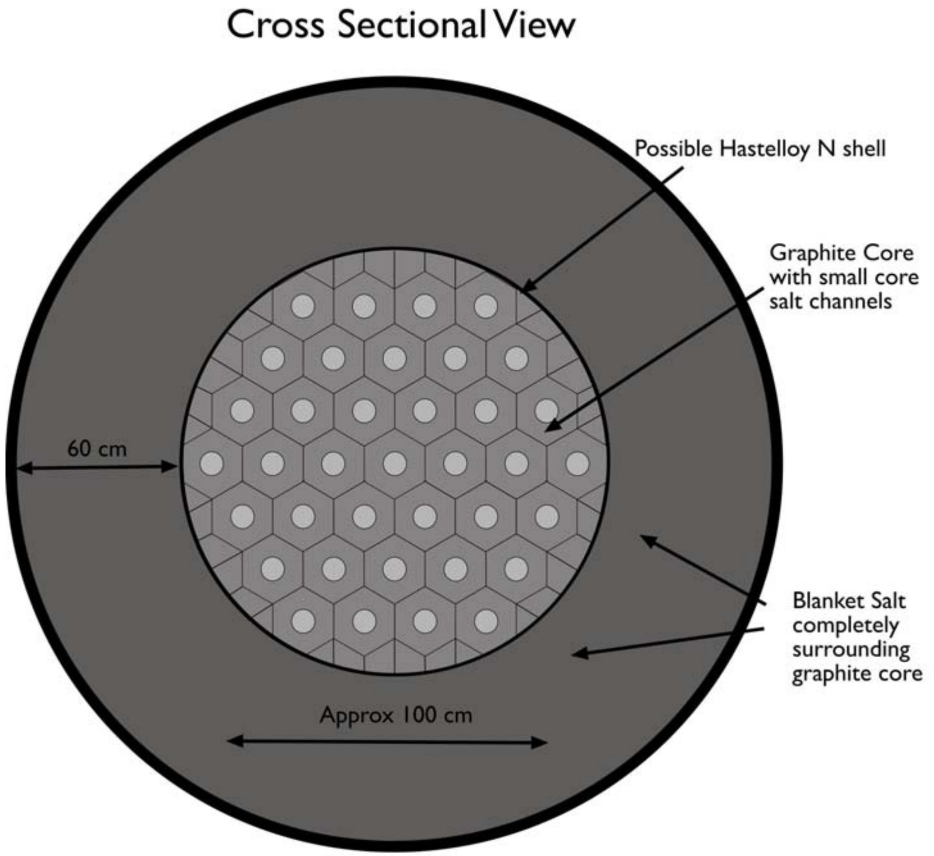



EXPANSION BELLOWS
GRAPHITE OR STEEL REFLECTOR
BLANKET SALT
BLANKET CALANDRIA WALL
CALANDRIA GAP WITH POSSIBLE INSULATION
CORE WALL
CORE SALT

Graphite Free Version



Cross Section with Graphite Moderator (Graphite Pebbles as Alternative)



System Advantages

- Can use simple 2 fluid fuel processing without the "plumbing problem"
- Very strongly negative fuel salt coefficients
- Blanket will **also** have negative temp/void coefficient as it acts as a partial reflector
- Ease of graphite core fabrication (and replacement if necessary)
- Ease of modeling and prototyping
- Fissile inventory of 400 kg per GWe or even lower is possible.

Example: Graphite Free, Carbon or SiC composite for barrier

- The ORNL 4 ft sphere case with 0.16% $^{233}UF_4$ should be able to reach Break Even
- This equates to 94 cm diameter in elongated cylindrical geometry
- Assuming;
  - Core power density of 200 kW/L
  - 2 m/s salt velocity in core
  - Typical 565 C/705 C for Inlet/Outlet Temp
- Gives **404 MWe** (911 MWth), 6.6 m core

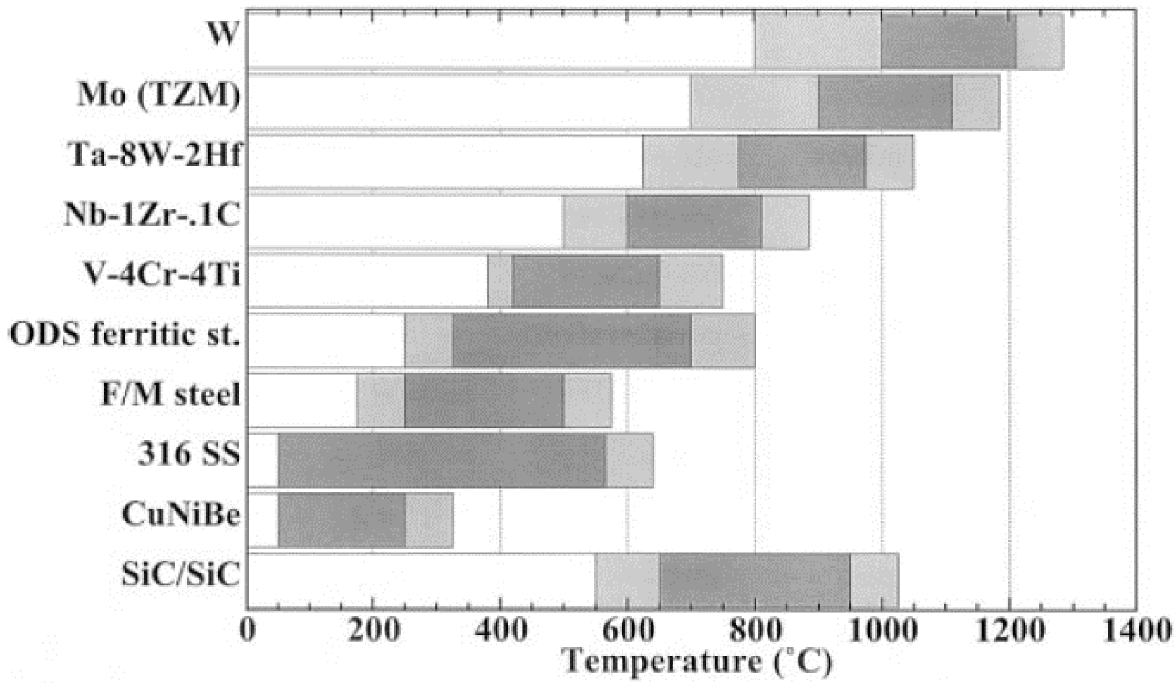
Other Variations

- Modestly higher concentration of $^{233}UF_4$ can allow:
  - Metal barriers such as Hastelloy N, Stainless Steels, Molybdenum
  - Alternate carrier salts to reduce costs and tritium production
  - Even greater simplification of fission product processing. 20 year or beyond removal time for fission products

Critical Issue: Core-Blanket Barrier

- Viability of barrier materials in high neutron flux
  - Much recent work in the fusion field using same $2^7LiF\text{-}BeF_2$ salt as coolant
  - SiC/SiC composites leading candidate
  - Hastelloy N and Stainless Steels possible with a modest temp reduction
  - Ease of "retubing" means even if limited lifetime still may be attractive

Fusion Barrier Candidates Studied



Operating temperature windows (based on radiation damage and thermal creep considerations)

"Operating Temperature Windows for Fusion Reactor structural Materials"
Zinkle and Ghoniem, 2000

Other Active Development Areas

- 1 and ½ Fluid version (thorium also in fuel salt)
  - Shifts design back toward "spherical" geometry
  - Can retain simple fuel processing by allowing minor "losses" of thorium
  - Barrier sees far lower neutron fluence
- Method for Startup on LEU
- Barrier Free Design (confidential)
- Lower Temp Operation (~550 C max)
  - Can allow use of steel instead of expensive Hasteloy N throughout circuit
  - Lifetime of any graphite greatly extended
  - Barrier choices vastly improved
- Simplified "Converter" Reactors

Startup on LEU

- Startup of the pure Th-^{233}U has always been a problem
  - ^{233}U produced elsewhere
  - HEU ^{235}U
  - TRUs or Pu alone
- Use of LEU ^{235}U deemed impractical as it takes decades to burn off ^{238}U

Method of Startup using LEU*

*Patent Pending

- Purchase LEU (<20% ^{235}U) for startup in fuel salt of desired 2 Fluid design
- As consumed, buy and add more LEU (no salt processing yet)
- Surrounding blanket salt will be producing approx 300 to 400 kg ^{233}U per GW year
- After only 1 to 5 years:
  - Stop reactor, fluorinate out and sell remaining LEU to other users
  - Remove blanket produced ^{233}U and start pure Th-^{233}U cycle with clean new carrier salt

Molten Salt Converter Reactors

- Starting Premise is Oak Ridge`s 30 year Once Through Design (1980)
  - 1000 MWe output
  - Startup with LEU (20% ^{235}U) + Th
  - No salt processing, just add LEU
  - Large, low power density core gives 30 year lifetime for graphite (10m x 10m)
  - Similar startup fissile load as LWR
  - Averages a Conversion Ratio above 0.8

Molten Salt Converter Reactors

- 1810 tonne Lifetime Ore Requirement
  - 5808 for LWR (3702 with Pu recycle)
  - 4456 for CANDU (2196 with Pu recycle)
- At end of 30 years
  - Uranium easily removed and reused
  - Transuranics recycle also possible
    - ~1 tonne TRUs in salt at shutdown
    - In that timeframe, LWR 8 tonnes to waste
    - CANDU about 12 tonnes
  - Fission Products can be concentrated by vacuum distillation or left in solidified salt

Possible Improvements
(at the expense of lower conversion ratios)

- Graphite Pebbles as moderator
  - Removes need for flux flattening
  - Can go to smaller, higher power core
  - Pyrolytic Coatings for increased safety
- Carrier salt switch
  - $NaF\text{-}BeF_2$ low cost, low melting point
  - $NaF\text{-}ZrF_4$ low cost, no tritium production
- Graphite Free "Tank of Salt" Core
  - Retain thermal spectrum by having very low fuel concentration and let the carrier salt act as moderator (Be, Li, F)

Possible AECL Interests

- If ACR 1000 does not proceed and supercritical water remains a “challenge”, how about a Plan C?
- Heavy Water moderated versions have been and continue to be contemplated worldwide
- At a minimum, the use of clean molten salt as a low pressure coolant in CANDU design would appear worth investigating

What Way Forward?

- Corporate interest will always be difficult to attract
  - No fuel fabrication contracts
  - Min 15 year return on investment a tough sell to shareholders (no matter how big the return may be)
  - Existing nuclear players have their choices in place

What Way Forward?

- Other Corporate Players?
  - Big Oil
    - For a small fraction of current profits, can retain their position in the energy market after “Peak Oil”
  - Chemical Giants
    - A majority of the needed R&D and engineering work would fit their skill set
- Individuals with Deep Pockets?
  - What better way for those such as Gates, Branson, Allen, Buffet to invest in the future

What Way Forward?

- International Cooperation
  - ITER model as guide but with greater corporate involvement
- Start Small, Think Big
  - Year 1 Create a short list of designs
  - Year 2 Choose 2 or 3 concepts to receive significant design funding
  - Year 3 to 8 Prototype design and construction
  - Year 5 to 13 Demonstration Reactor design and construction (of 1 or more concepts)
  - Year 15 to 20 Add several units with modifications as needed
  - *Year 20+ Build them by the thousands...*