Liquid Fluoride Reactors: A Luxury of Choice

Oct 20th 2009

Thorium Energy Alliance Conference

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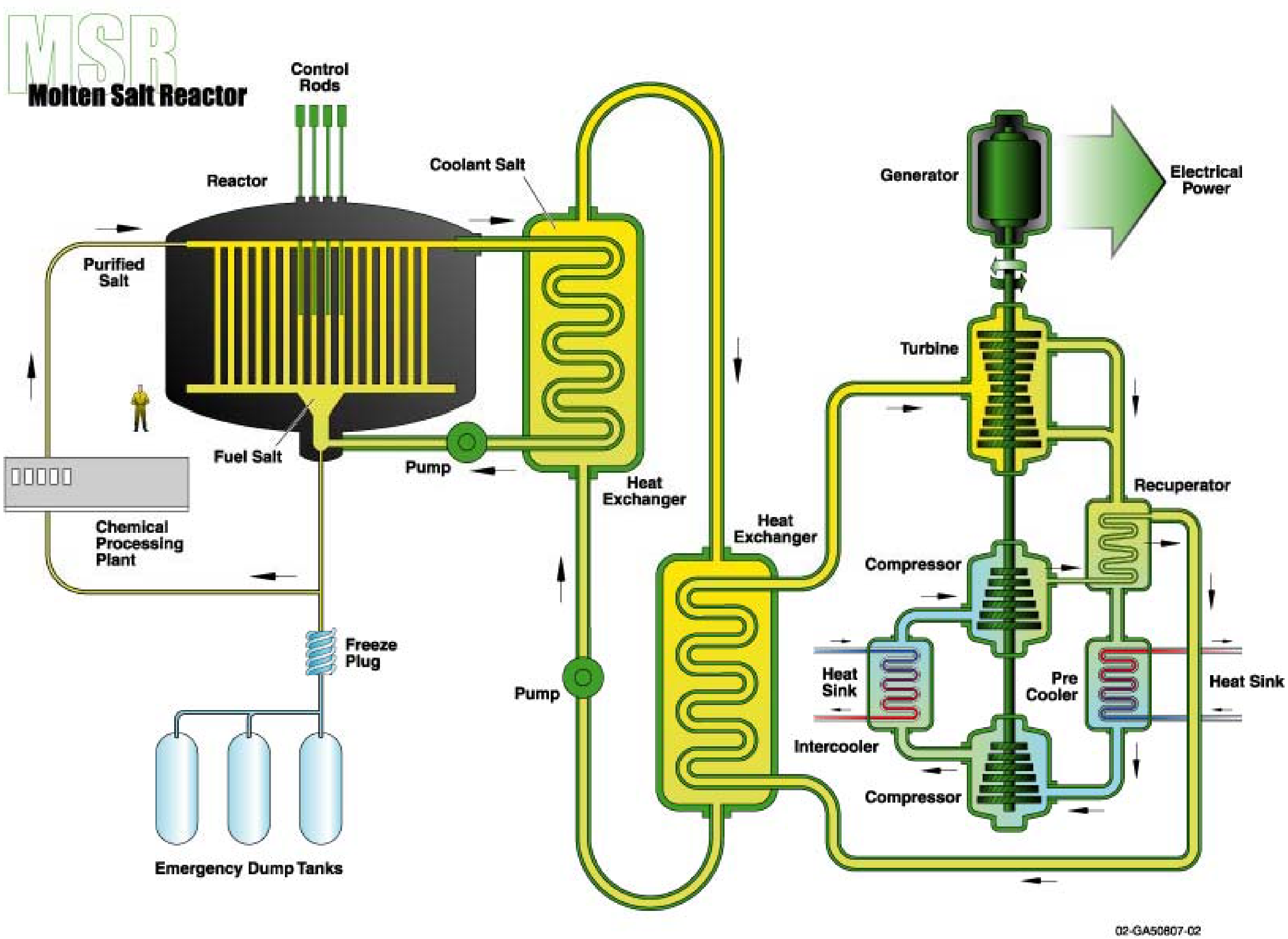
The Single Fluid, Graphite Moderated Molten Salt Breeder Reactor (MSBR)



General Attributes of the "Traditional" 1000 MWe Single Fluid MSBR Design

- $2^{7}LiF-BeF_2$ - 12% ThF_4 - 0.3% $^{233}UF_4$
- Hastelloy N used for piping, HX, vessel
- 565 ^{o}C Inlet, 700 ^{o}C Outlet Temp
- 44% on Steam to 48% on Gas
- Fission Products removed 20 day cycle
- ^{233}Pa removed on 10 day cycle
- Startup Fissile Inventory of 1500 kg
- Breeding Ratio of 1.06
- 20 year doubling time

General Benefits of Liquid Fluoride Reactors

- Volatile fission products including Xenon simply bubble out, are removed and stored elsewhere. A spill will simply solidify
- Salts have high boiling point and operate at low pressure
- A temp increase instantly lowers reactivity
- No possible steam explosions, hydrogen production. No need for the containment building to hold in steam like LWRs
- Control rods or burnable poisons not required. Almost no excess reactivity
- A simple freeze plug melts upon fuel overheating to drain the salt to critically safe, passively cooled dump tanks

Liquid Fluoride Fundamentals
Single Fluid vs Two Fluid

- Single Fluid obviously is only one fluid carrying both fissile and fertile
- Two Fluid design has a separate salts for $^{233}UF_4$ and ThF_4
  - Improves reactivity coefficients
  - Simplifies fission product removal
  - Can avoid need for ^{233}Pa removal
  - Complicates core design
    - *Not as much as thought!*
- 1 ½ Fluid design, mix of both types

ORNL's Two Fluid Reactor
Mid 1960s focus

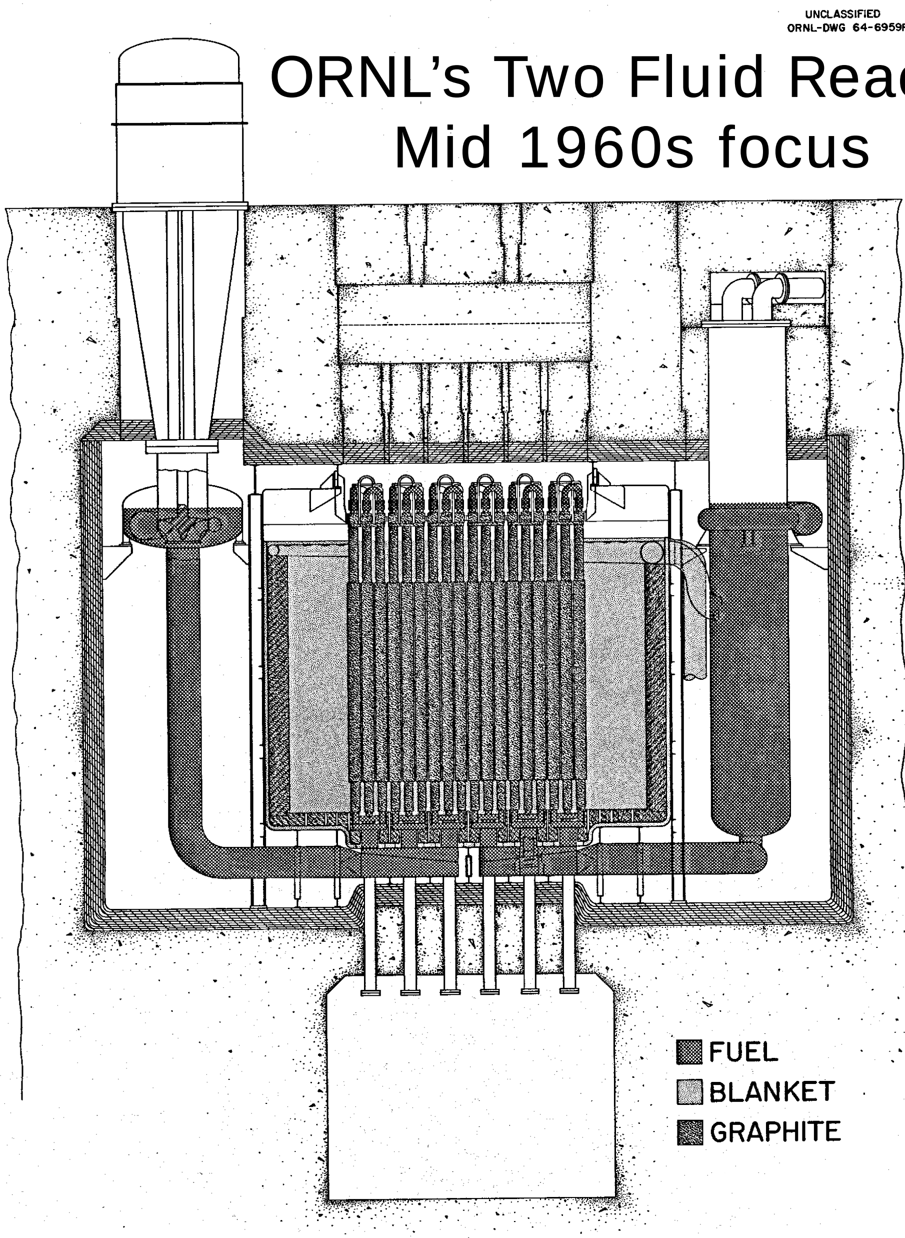
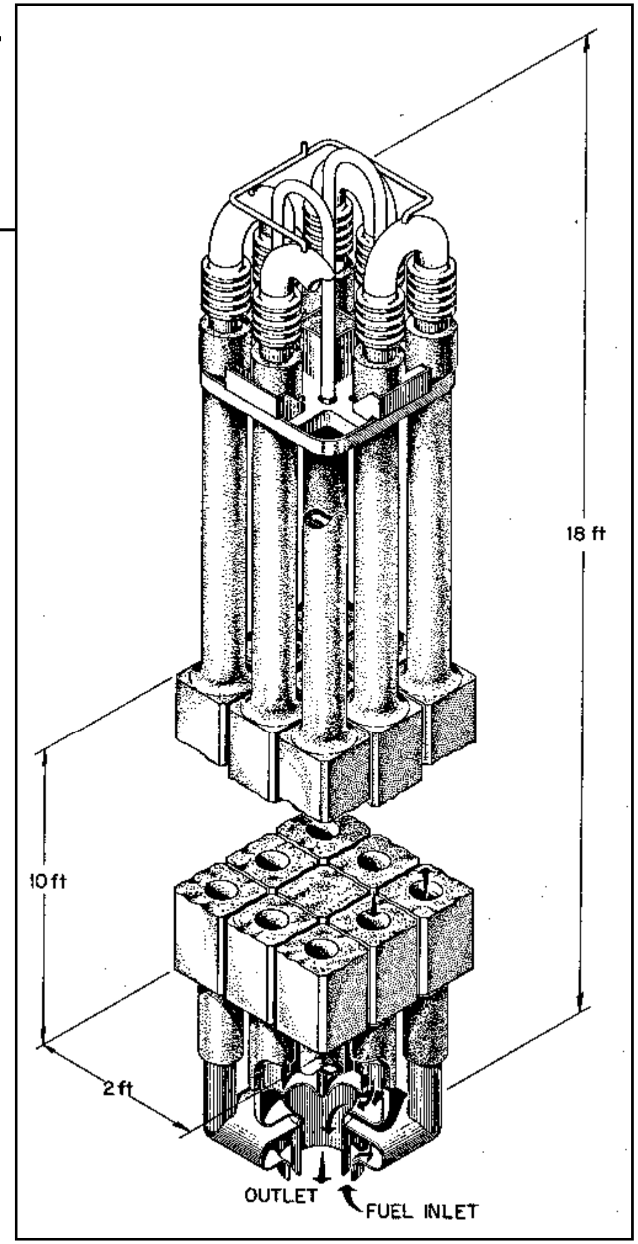


Fig. 2. Two-Region Molten-Salt Breeder.



Great concept except for the "plumbing"!

Liquid Fluoride Fundamentals
Fuel Processing

- Uranium removed by bubbling fluorine gas through the salt to convert UF_4 to UF_6
- ^{233}Pa may require removal ***some designs***
- Stable fission products, most importantly the rare earths, can then be removed
  - Liquid Bismuth Reductive Extraction
  - Vacuum Distillation (only Two Fluid designs)
- Thorium is chemically similar to rare earths and complicates FP removal
- Fission product removal can range from 10 days to several years ***OR not at all***

Liquid Fluoride Fundamentals
Graphite Behaviour

- Graphite often used as a moderator
- Very compatible with the liquid salts
- As with any reactor use of graphite, it will have a limited lifetime due to fast neutron damage (it first shrinks, then swells)

Liquid Fluoride Fundamentals
Long Term Waste Reduction

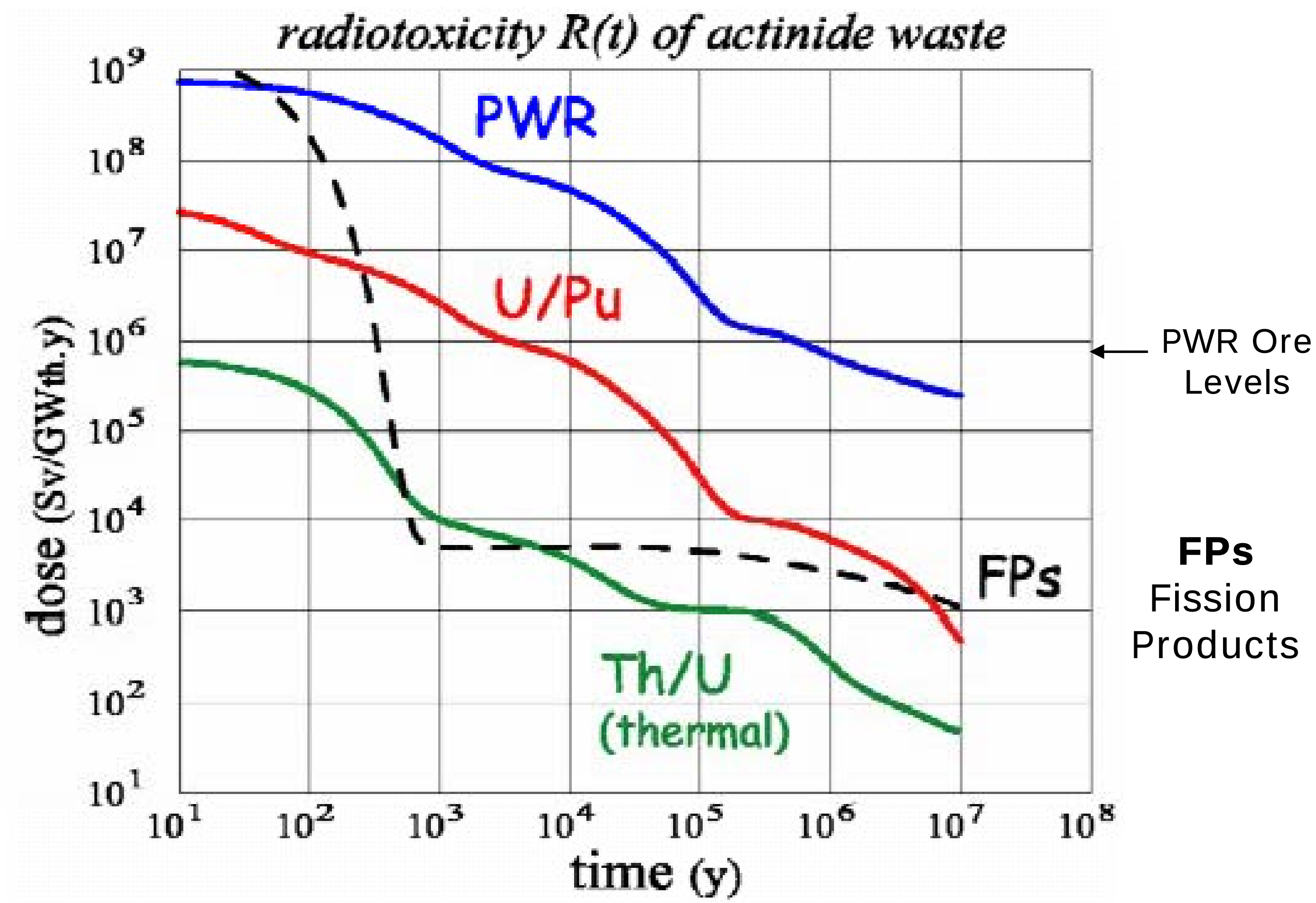
- The ***long lived*** radiotoxicity of spent fuel is dominated by transuranic elements Pu,Np,Am,Cm

- Minimizing TRU waste turns long term storage/disposal into a few hundred year job and removes recriticality issues

Radiotoxicity PWR vs FBR* vs LFR*

*Assuming 0.1% Loss During Processing

Data and graph from Sylvain David, ***Institut de Physique Nucléaire d'Orsay***



Turns waste management into 500 year job, not million year

Proliferation Resistance
The *Pure* Thorium – ^{233}U Cycle

- ^{232}U present in significant quantities
  - 69 year half-life with strong 2.6 MeV gamma ray from daughter product ^{208}Tl
  - Makes illicit use difficult and highly detectable
  - No national program ever based on ^{233}U
- ^{233}U can be ***instantly denatured*** by dumping $^{238}UF_4$ into the molten fuel salt
- ^{233}Pa removal can lead to “clean” ^{233}U and thus should be avoided
- Only small amounts Plutonium are present, it is of poor quality and very hard to extract

Proliferation Resistance
Denatured Cycles

- The pure Th-^{233}U cycle does represent the use of Highly Enriched Uranium
- Running denatured by including enough ^{238}U makes any uranium useless for weapons
- It does mean more plutonium present but still of poor quality and very hard to remove
- Including ^{238}U will typically lower conversion ratios

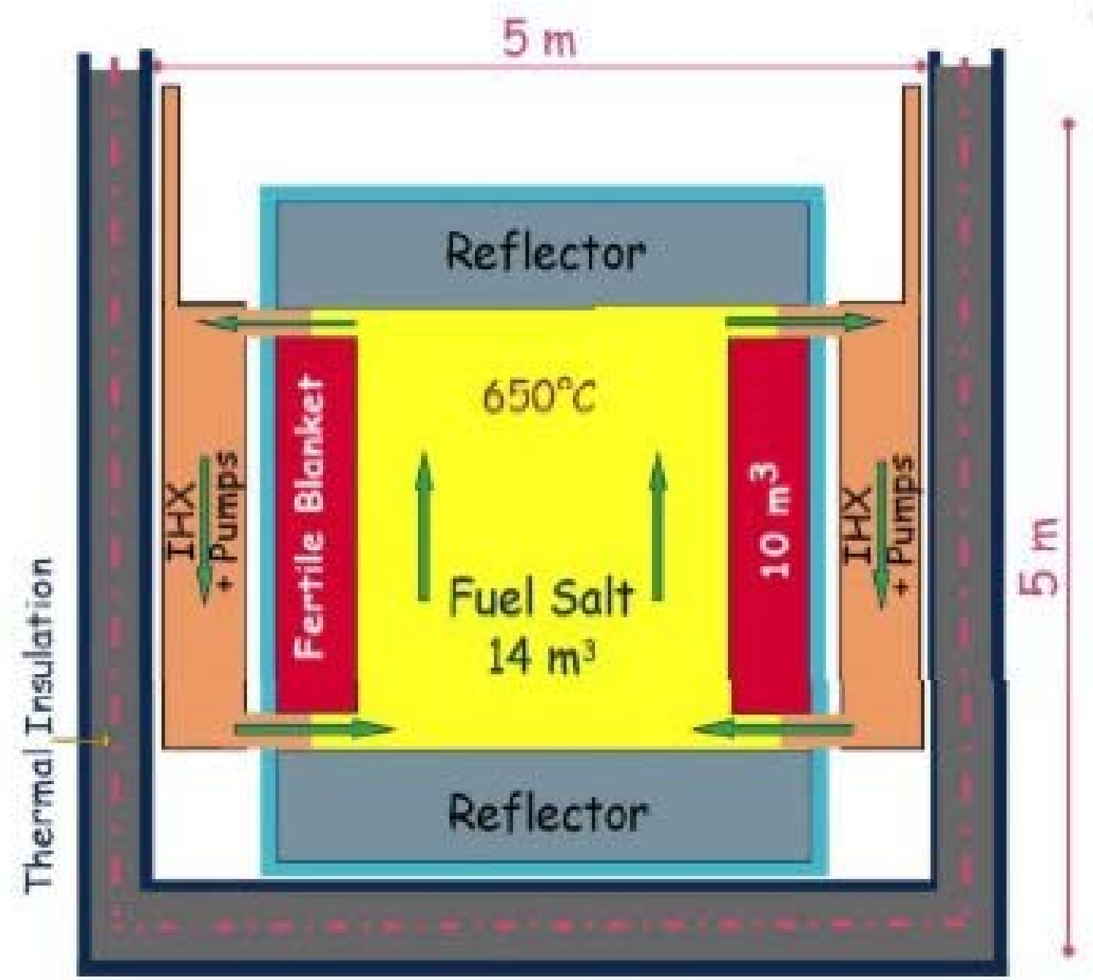
History of the MSR program at Oak Ridge National Laboratories

- Aircraft Reactor Program ***1950s***
- Sphere within Sphere Two Zone power reactors ***late 1950s***
- Two Fluid Graphite ***1960 to 1968***
- MSRE test reactor ***1965 to 1969***
- Single Fluid Graphite ***1968 to 1972***
- Major Funding Cut ***1972***
- Work on Denatured designs ***late 1970s***

A Resurgence of Interest

- Chosen as one of six Gen IV designs
- Major program in France
  - TMSR removes graphite moderation
  - Has radial Th blanket salt
    - Makes it a partial 1 ½ Fluid design
  - Harder spectrum
  - 5.5 tonnes ^{233}U for startup
  - Upwards of 1.13 Breeding ratio
  - Minimal salt volume of 20 m^3

The French TMSR
Thorium Molten Salt Reactor



Design has a thorium blanket but only radial, not axially (which would be very difficult)

A Resurgence of Interest

- Extensive work also done in the Czech Republic with an expertise in fluoride chemistry
- Strong program in Russia, focused on burning transuranics in MOSART design
- Numerous groups worldwide such as Japan, Netherlands and Canada
- U.S. efforts have mostly shifted towards Liquid Fluoride “Cooled” designs as an intermediate step. Much possible overlap of R&D

Back to Basics

- What ways can we make a good design even better
- Several areas that could be improved
  - Reactivity coefficients with graphite
  - Cost, complexity and “image” of fuel processing
  - Real and/or Imagined Proliferation issues of the pure Th – ^{233}U cycle

New Ideas for Consideration

- Reviving the Two Fluid Concept
  - A simple solution to the “plumbing problem”
- Simplified Converter Reactor
  - Expanding on ORNL`s “30 Year Once Through” design

Solving the Two Fluid Plumbing Problem

- Sphere-Within-Sphere with a single barrier is simplest approach
- BUT, if inner fuel salt lacks thorium then the critical diameter must remain small (approx 1 m with or without graphite)
- Too small for power plants
- “Standard” conclusion was fuel and blanket salts must be interlaced by graphite or metal plumbing

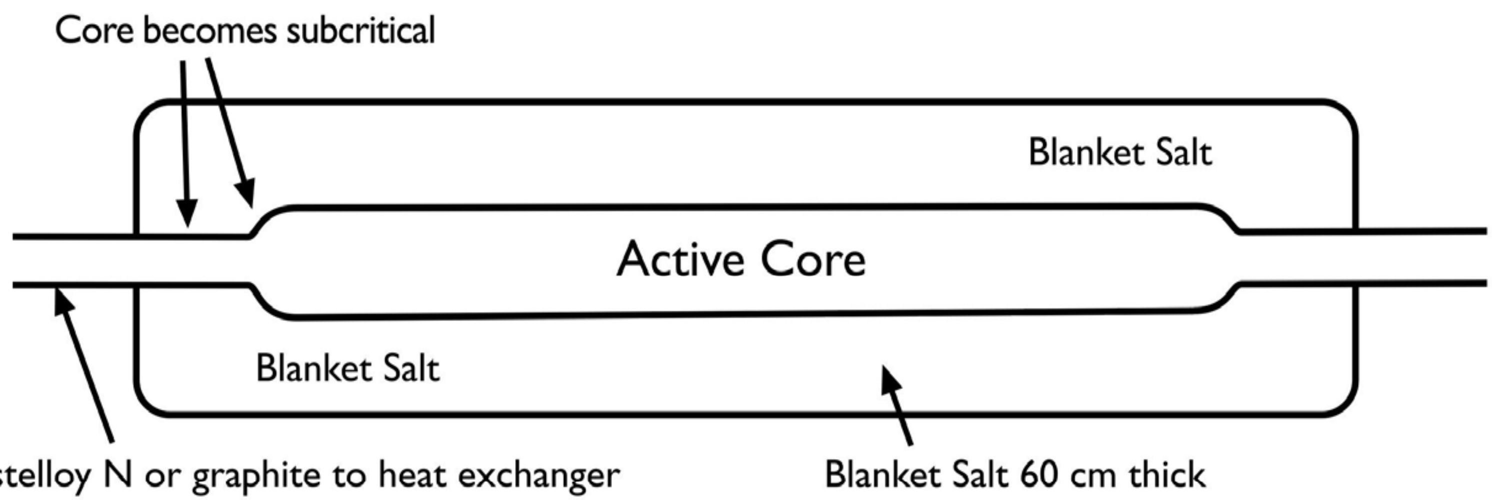
Modified Geometry Two Fluid Reactor "Tube-Within-Shell"

*Patent Pending

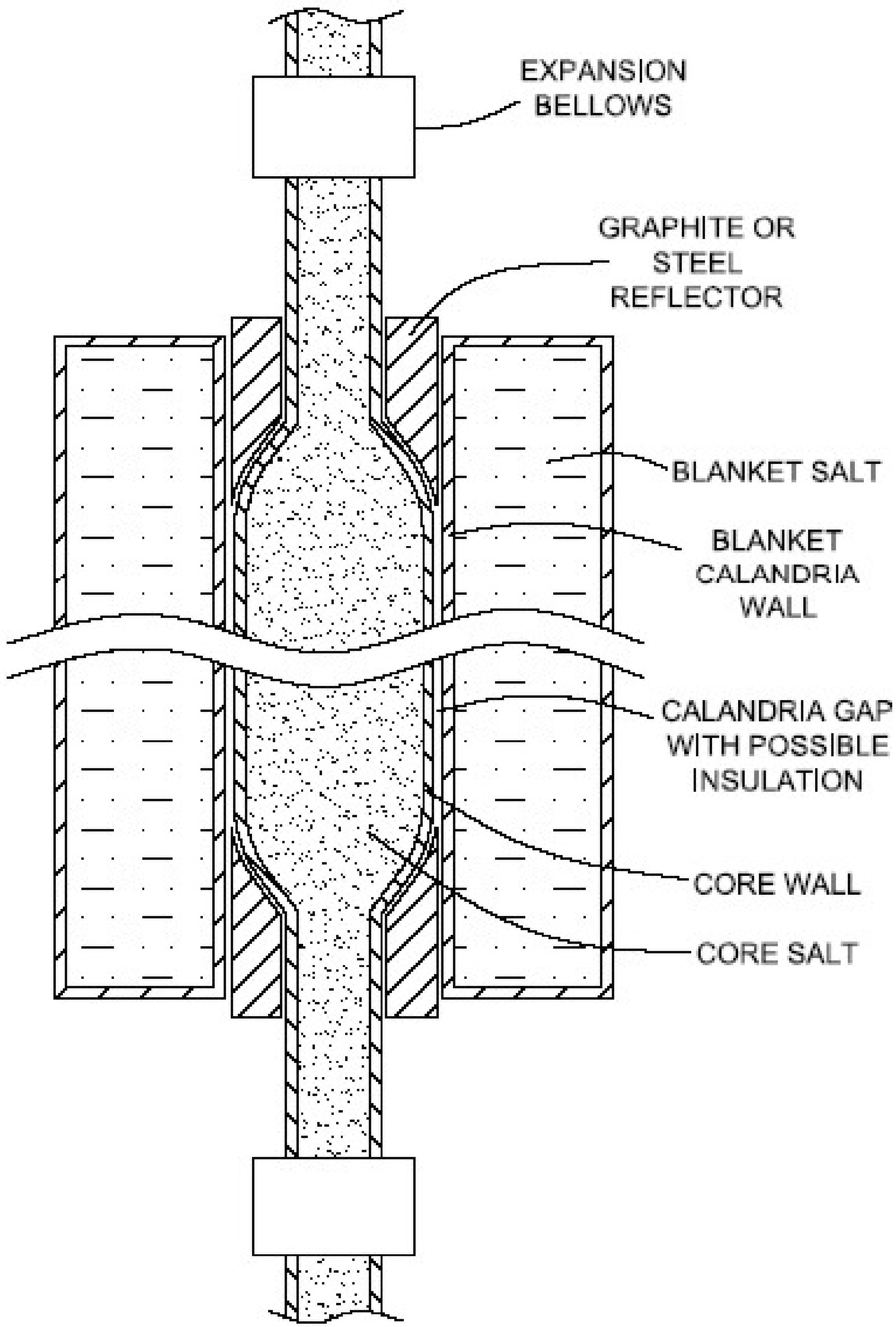
Side View of Reactor Core and Surrounding Blanket Salt

Core is Graphite + Fuel Salt or 100% Fuel Salt

Typical Diameter of 1 meter



Expands power producing volume while maintaining the small inner core needed for a simple 2 Fluid design


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

New Concepts Advantages

- Can use simple Two Fluid fuel processing without the “plumbing problem”
- Excellent reactivity coefficients for both the fuel and blanket salts
- No need for graphite in the core
- Simple, transportable cores
- Ease of modeling and prototyping
- Fissile inventory of **400 kg per GWe** or even lower is possible
- Key issue is confirming a barrier material between the core and blanket

Simplified Liquid Fluoride Converter Reactors

- Starting Premise is Oak Ridge`s *30 Year Once Through Design* (1980)
  - Single Fluid, 1000 MWe output
  - Startup with LEU (20% ^{235}U) + Th
  - No salt processing, just add small amounts of LEU annually
  - Low power density core gives 30 year lifetime for graphite (8m x 8m)
  - Lower fissile startup load than LWR (3450 kg/GWe)
  - Better reactivity coefficients than MSBR
  - Almost no R&D needed to build now

Denatured Molten Salt Reactors

- 1800 tonne Lifetime U Ore Requirement
  - 6400 tonnes for LWR
  - 4900 tonnes for CANDU
- At end of 30 years
  - Uranium easily removed and reused which would drop the lifetime ore to under **1000 t**
  - Transuranics should also be recycled
  - ~1000 kg TRUs in salt at shutdown
  - Assuming typical 0.1% processing loss, just 1 kg in 30 years! Same low long term radiotoxicity as pure Th-^{233}U cycle

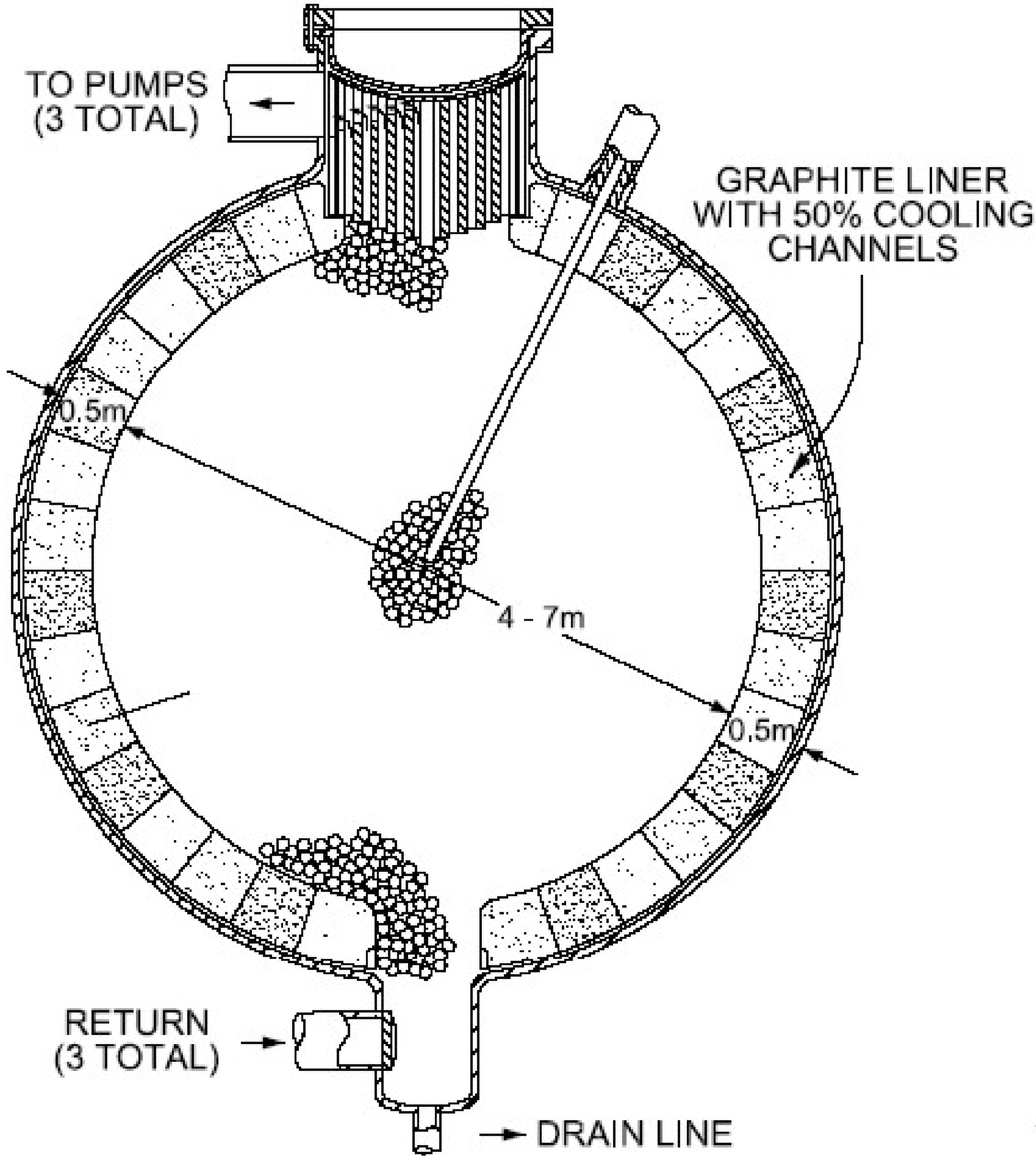
Ultimate in Proliferation Resistance

- No fuel processing ever required
- Uranium is never usable for weapons
- Any Pu present is of very low quality, very dilute in a highly radioactive salt and very hard to remove from the salt
- No way to cycle in and out fertile to produce fissile like solid fuel reactors

New Suggested Improvements

- Higher power density, smaller cores
  - Halve the volume and replace graphite and/or salt every 15 y
  - Much lower starting inventory and better uranium utilization
  - Minor increase in neutron losses to ^{233}Pa

- Graphite Pebbles as moderator
  - Simple to manufacture and replace
  - Can go to smaller, higher power cores
  - Pyrolytic coatings for increased safety



Proposed Pebble Bed DMSR Converter

Suggested Improvements

- Graphite Free "Tank of Salt" Core
  - Difficult due to ^{238}U resonances unless high fissile inventory employed
- Alternate carrier salts
  - **NaF-BeF$_2$** low cost, low melting point
    - May allow stainless steel throughout loop
  - **NaF-RbF** low cost, no tritium production
    - Simplification of entire primary loop
  - **NaF** very low cost, higher melting point
    - These alternatives will only roughly double the neutrons absorbed in the salt from 1.5% to 3%

Reactor	Lifetime Uranium Ore (t)	Annual Uranium Ore (t)	Annual Ore Costs **50$/kg U**	Annual Fuel Costs **50$/kg U**	Annual Fuel Costs **5000$/kg U**
LWR	**6400**	200	8.5 million	~40	~880
LWR with U-Pu Recycle	**4080**	125	5.3		
Sodium Fast Breeder	**2400** **If start up on ^{235}U**	1			
DMSR Converter	**1800**	35	1.5	~6 0.001$/kwh	~155 <0.02$/kwh
DMSR single U recycle	**1000**	35	1.5	~6	~155

Based on 0.2% tails, 75% capacity factor, 30 year lifetime

LWR data from "A Guidebook to Nuclear Reactors" A. Nero 1979

3.9 million$ annual enrichment costs for DMSR at 110$/SWU

At $5000/kg, uranium from sea water likely feasible and unlimited resource

Peak Uranium?

- Attractiveness of converter designs hinges on sustainability of uranium resources and mining
- As shown however, even an enormous price increase has little effect on fuel costs for a DMSR which assures resources for a very large fleet of DMSRs

Conclusions

- Molten Salt designs have inherent features that favour overall safety, waste reduction, low cost and rapid deployment
- They also have great flexibility to match varying priorities
  - Can attain the absolute highest levels of proliferation resistance
  - Can run on minute amounts of thorium, or modest amounts of uranium for the utmost in simplicity

Backup Slides

Seven Reactor Design Priorities

- Safety
- Overall Power Costs
  - Capital Costs
  - Fuel plus O&M
- Long Term Waste Radiotoxicity
- Resource Utilization
  - Quantity and Quality of Starting Load
  - Annual Requirements
- R&D Requirements
- Capability for Rapid Deployment
- Proliferation Resistance
  - Subversion to a State Program
  - Theft and/or Terrorist Take Over

1950s and 1960s Design Priorities

- Safety – No problem...
  - If we engineer it right, do proper maintenance and extensively train our staff "There is NO safety issue"
- Power Costs – Important
- Resources – Extremely Important
  - We will run out of uranium by the 1980s
  - LWRs OK for now but we will need breeder reactors
- Rapid Deployment – Important
  - Power needs expected to continue to rise exponentially so breeder reactors must have very short doubling times

1950s and 1960s Design Priorities

- Proliferation Resistance
  - What?
- Long Term Radiotoxicity
  - What?
- R&D Requirements
  - Every concept needs plenty but funding is plentiful

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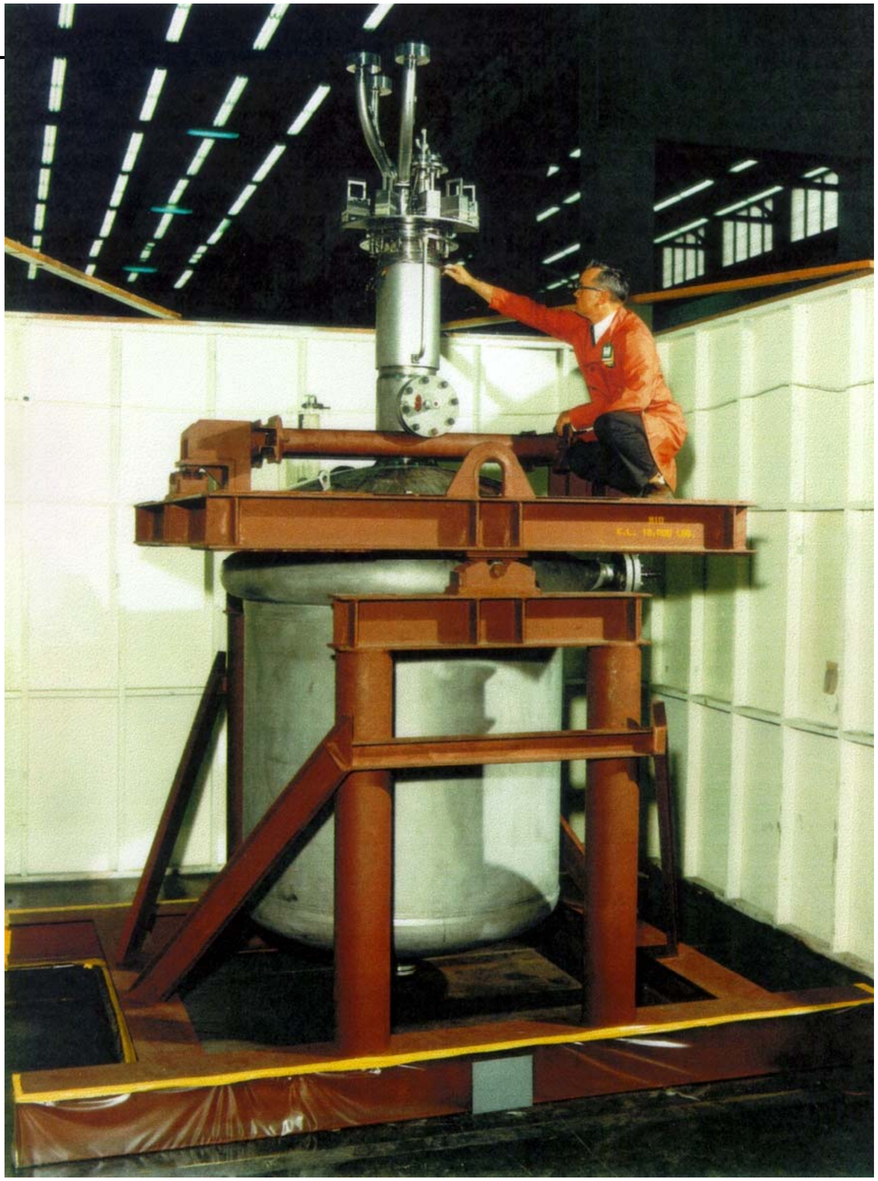
Molten Salt Reactor Experiment 8 MWth



A Strange Beginning
An Aircraft Reactor?

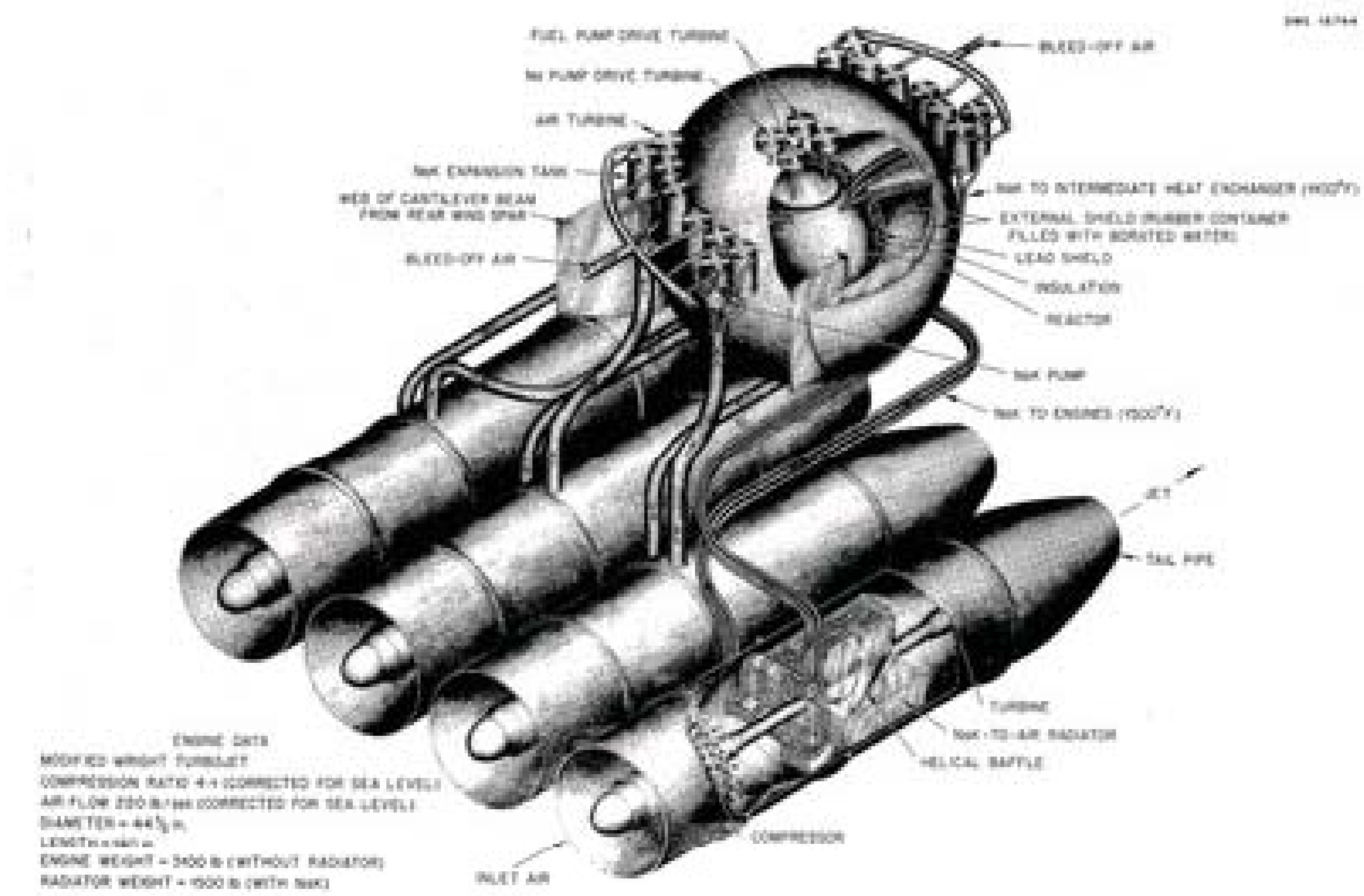
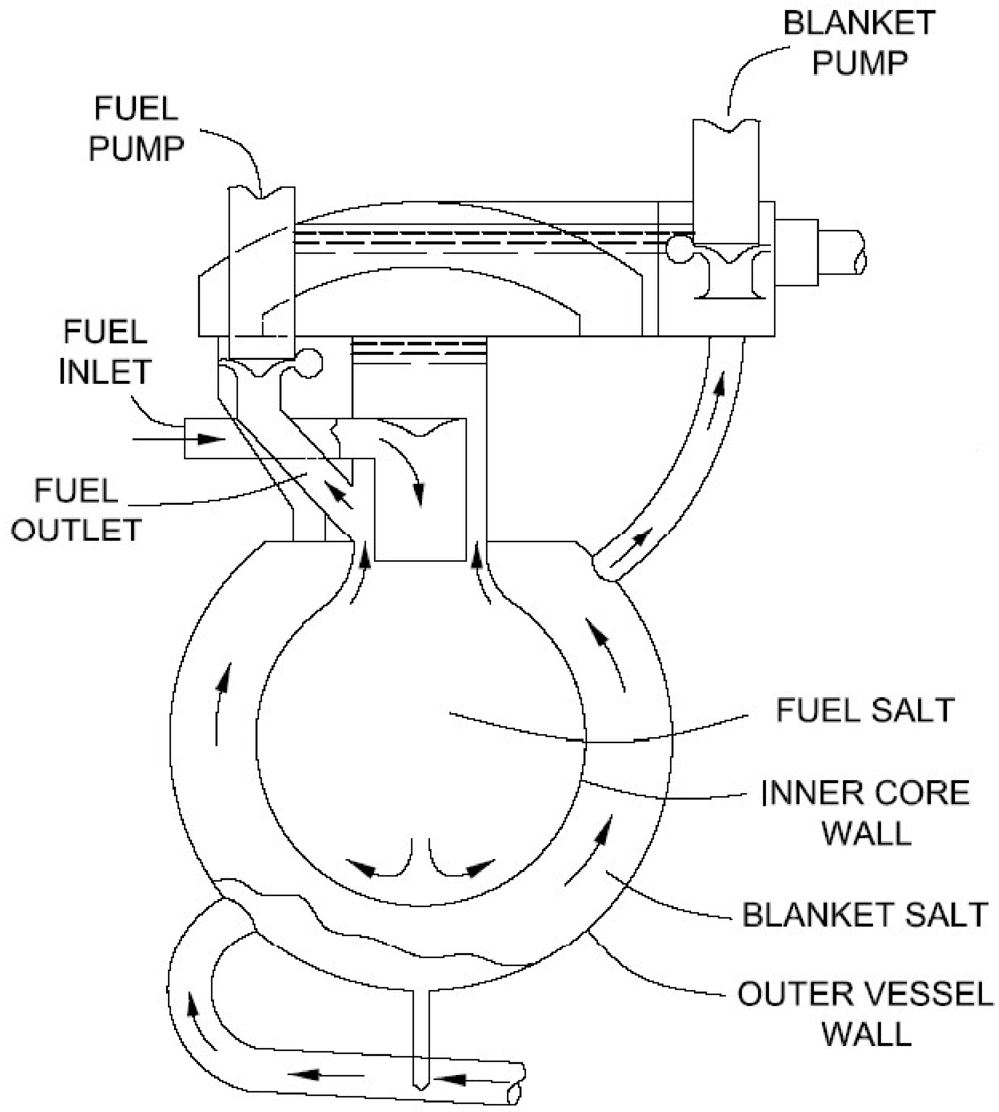


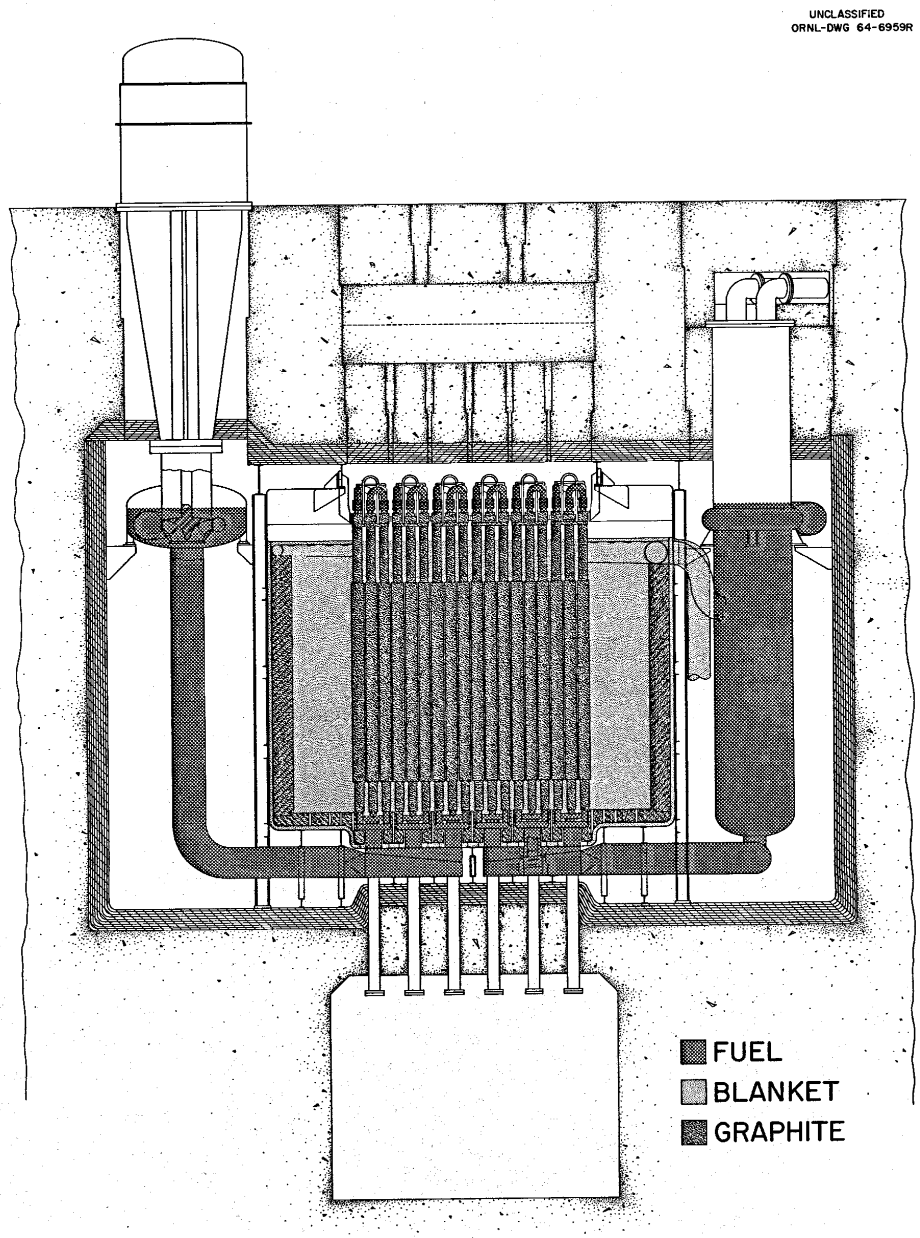
Fig. 4.33. Aircraft Power Plant (200 Megawatt).

The Aircraft Reactor Experiment

- Test reactor of early to mid 1950s
- Very high temperature 860 ^{o}C
- Canned BeO moderator
- $NaF-ZrF_4$ carrier salt
- Points the way to possible power reactors *(even if the idea of an airborne reactor far fetched)*

Homogenous Molten Salt Reactor
Late 50s ORNL





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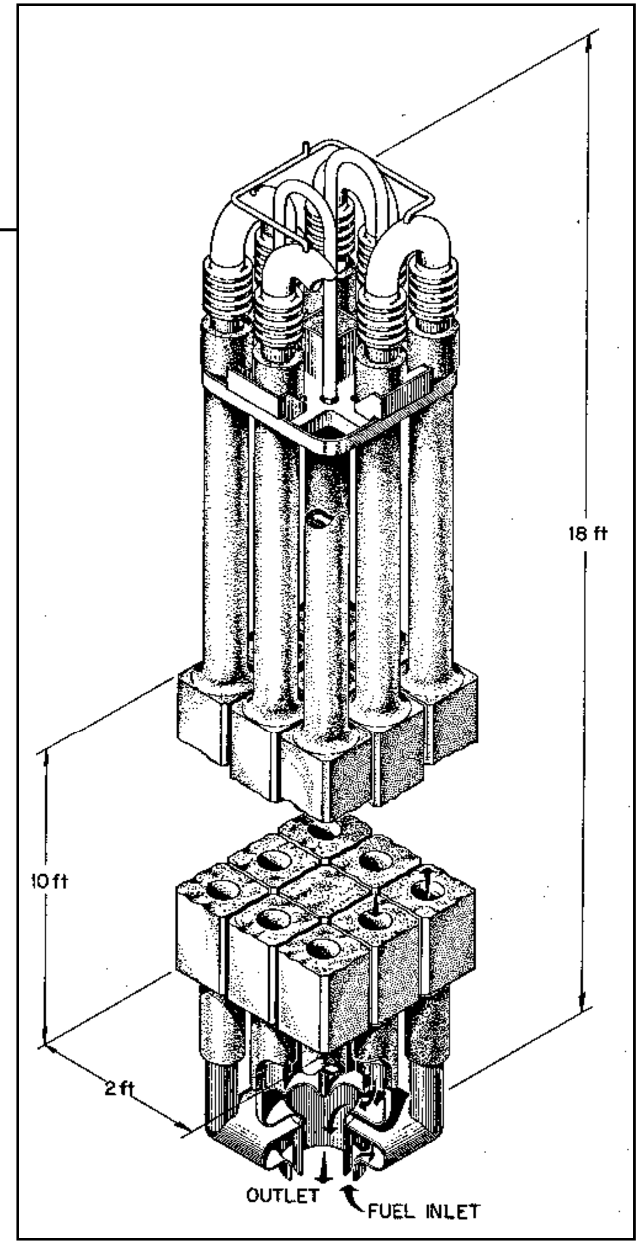


Quality of Produced Plutonium

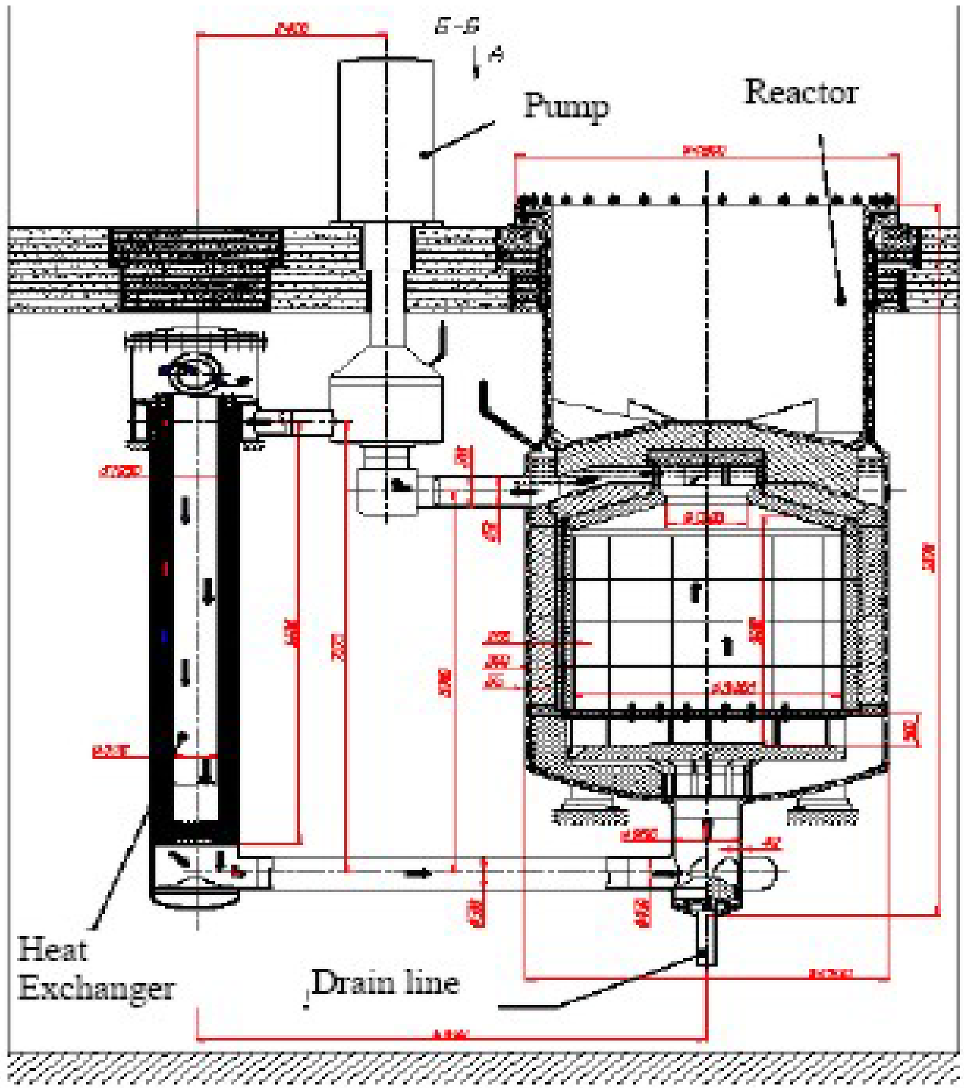
Isotope	Proliferation properties	PWR Reactor Grade	DMSR 30 Year Once Through	MSBR Pure Th – ^{233}U cycle
^{238}Pu	Generates heat from alpha emission	1.3%	12.6%	73%
^{239}Pu	Main fissile Component	60.3%	31.1%	9.5%
^{240}Pu	Spontaneous fissions high	24.3%	18.1%	4.4%
^{241}Pu	Fissile and adds hard gamma rays	5.6%	13.6%	4.8%
^{242}Pu	fertile	5.2%	24.3%	7.4%

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

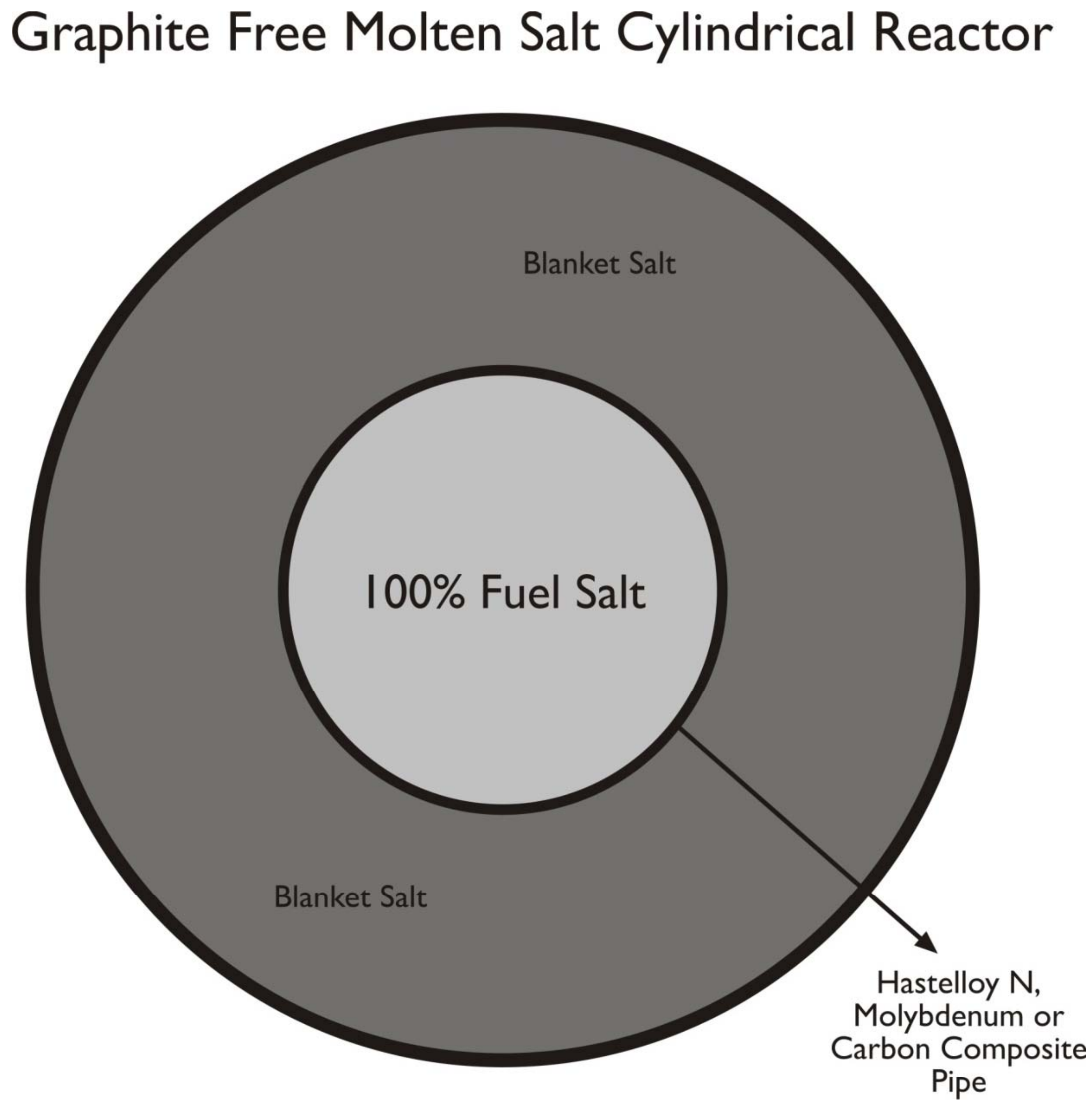
MSRE 8 MW(th) Reactor

- 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 found and mostly resolved in later years

Russian MOlten Salt Actinide Recycler and Transmuter   MOSART



Cross Section Graphite Free

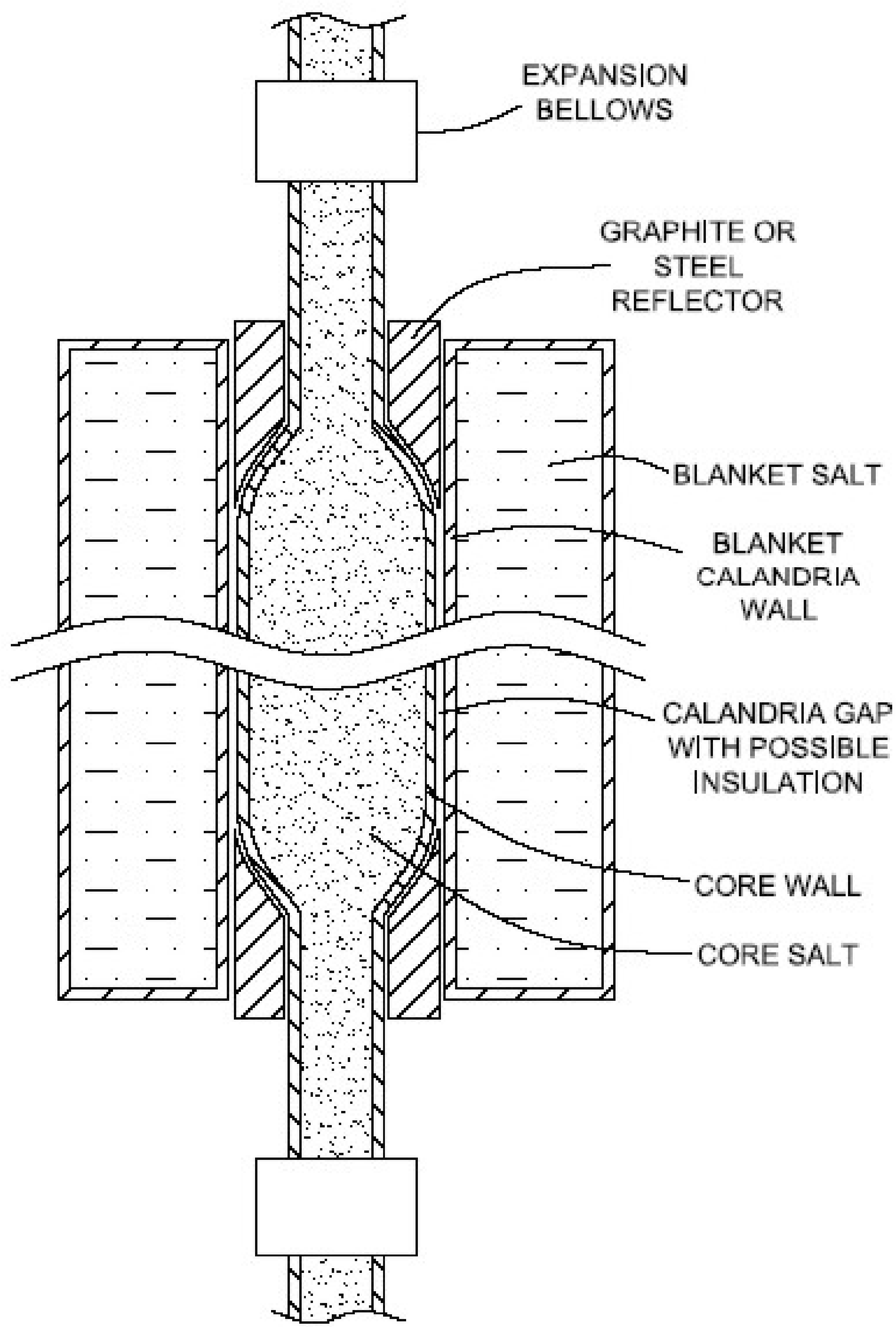


Example: Graphite Free, Carbon or SiC composite for barrier

- Using ORNL modeling for a 122 cm "spherical" core, 0.16% $^{233}UF_4$ should be able to reach Break Even Breeding
- A 122 cm sphere equates to 94 cm diameter in elongated cylindrical geometry
- Assuming;
  - Core power density of 200 kW/L
  - 2 m/s salt velocity in core
  - Standard 565 C/705 C for Inlet/Outlet Temp
- Gives **404 MWe** (911 MWth), 6.6 m core

Other Variations

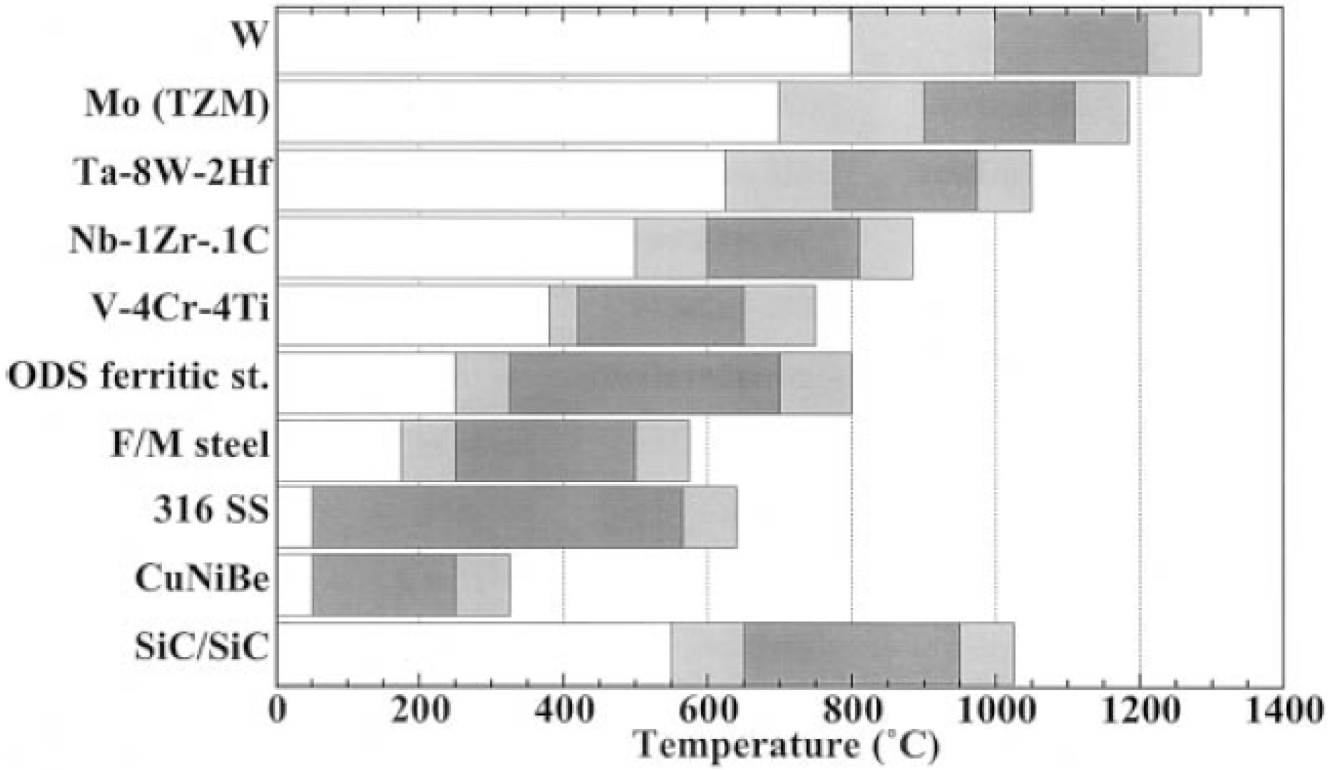
- Modestly higher concentration of $^{233}UF_4$ (0.2% to 1%) gives excess neutrons to allow:
  - Metal barriers such as Hastelloy N, Stainless Steels, Molybdenum
  - Alternate carrier salts to reduce costs and tritium production
  - Even greater simplification of fission product processing. 20 year or longer removal time for fission products


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

Critical Issue: Core-Blanket Barrier

- Viability of barrier materials in high neutron flux
  - Much recent work in the fusion field using same 2^7LiF-BeF_2 salt as coolant
  - Molybdenum, SiC/SiC or simple carbon composites leading candidates
  - Hastelloy N and Stainless Steels possible with a modest temp reduction
  - Ease of “retubing” means even a limited lifetime still may be attractive

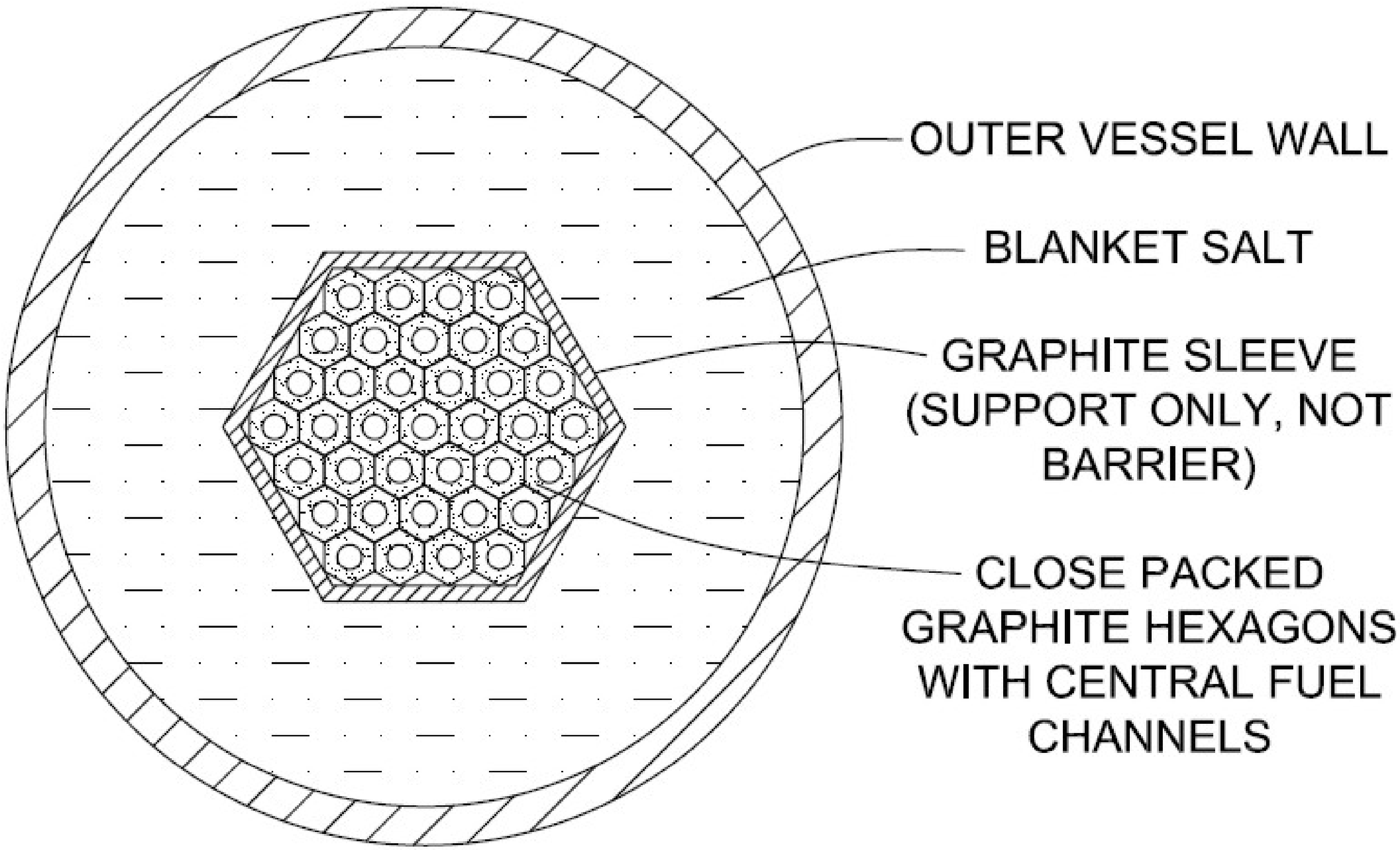
Fusion Structural Materials Studied

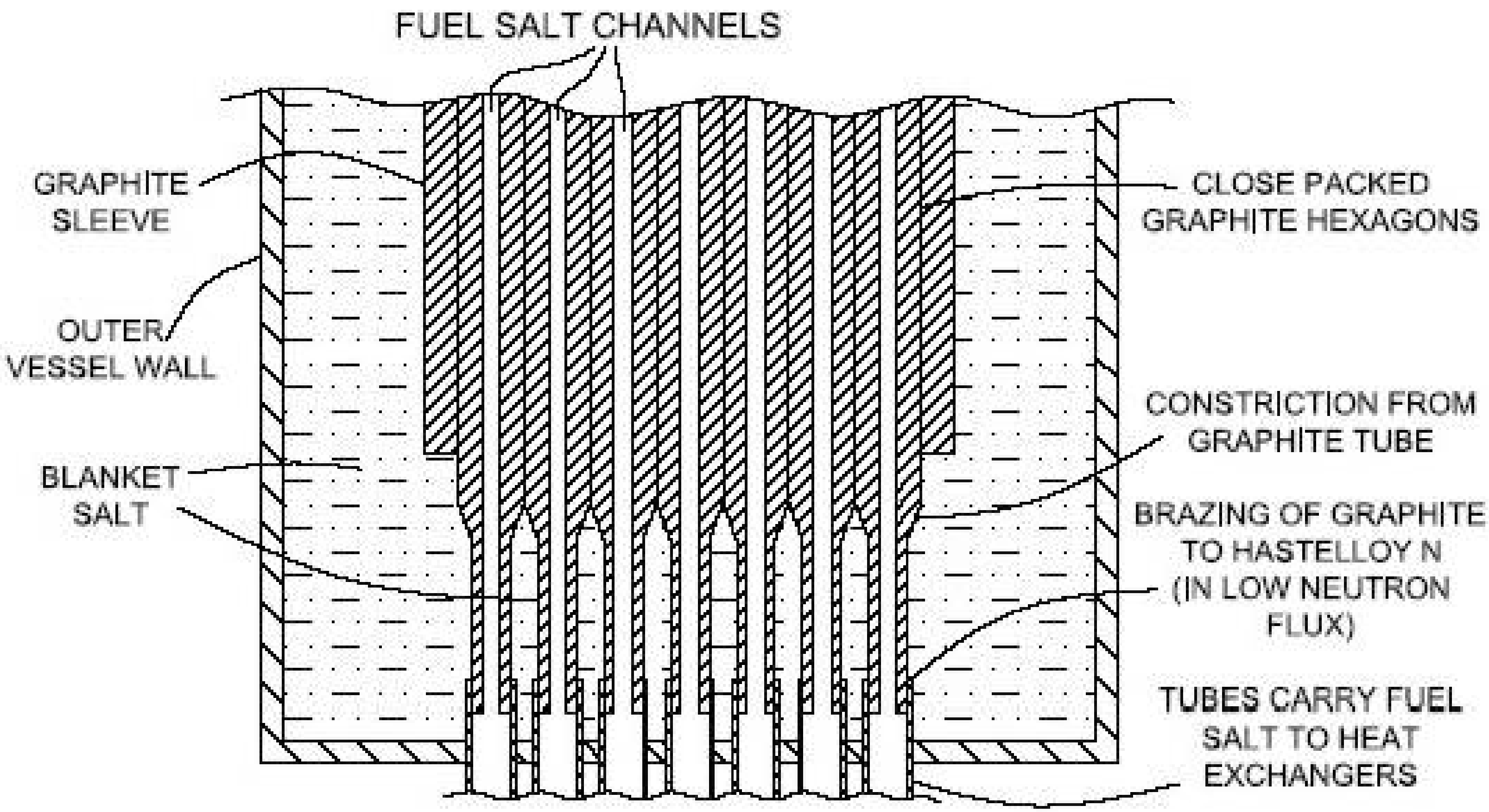


Operating temperature windows (based on radiation damage and thermal creep considerations)

"Operating Temperature Windows for Fusion Reactor structural Materials"

Zinkle and Ghoniem, 2000


OUTER VESSEL WALL
BLANKET SALT
GRAPHITE SLEEVE (SUPPORT ONLY, NOT BARRIER)
CLOSE PACKED GRAPHITE HEXAGONS WITH CENTRAL FUEL CHANNELS


FUEL SALT CHANNELS
GRAPHITE SLEEVE
CLOSE PACKED GRAPHITE HEXAGONS
OUTER VESSEL WALL
CONSTRICTION FROM GRAPHITE TUBE
BLANKET SALT
BRAZING OF GRAPHITE TO HASTELLOY N (IN LOW NEUTRON FLUX)
TUBES CARRY FUEL SALT TO HEAT EXCHANGERS

What Way Forward?

- Corporate interest will always be difficult to attract
  - No lucrative 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 is key way to spread the costs and rewards
- ITER model as rough guide but with greater corporate involvement
- Likely no one design will be best for all nations or utilities so best to move forward on several versions
  - 95% of R&D needed would serve entire community
  - Nothing like competition to yield the best results

What is Needed Short Term

- Neutronic modeling
- Fuel Salt chemistry and corrosion studies of various carrier salts and materials for heat exchangers or potential 2 Fluid barriers
- Non-nuclear component testing of pumps, valves, heat exchangers etc.
- Minor levels of funding to support these efforts (the hardest part of all!)

Two Region Homogeneous Reactor

Projected breeding ratios assume thicker blanket and alternate barrier. From ORNL 2751, 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*