Molten Salt Converter Reactors: From DMSR to SmAHTTR

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Conference on Molten Salts in Nuclear Technology

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The 1970s Single Fluid, Graphite Moderated Molten Salt Breeder Reactor (MSBR)
Advantages of Molten Salt Reactors

- **Safety**
  - Inherent safety, understandable to the public
  - Hard to even imagine accidents hazardous to the public

- **Reduced Capital Cost**
  - Low pressure, high thermal efficiency and far superior coolants (smaller pumps, heat exchangers)

- **Long Lived Waste Profile**
  - Even converter designs can have closed cycles that see almost no transuranics going to waste
  - Ideal system for consuming existing transuranic wastes

- **Resource Sustainability and Low Fuel Cycle Cost**
  - Thorium breeders obvious but MSR converters also extremely efficient on uranium use
Some Design Constraints

- ThF$_4$ and UF$_4$ easy to add to carrier salts, PuF$_3$ often very limited solubility
- High temperature limit set by materials (700 C conventional), lower limit by salt melting point, viscosity and PuF$_3$ solubility
- Graphite limited lifetime
  - Larger, lower power density for 30 y+ lifetime
  - Higher power density must periodically replace graphite
Design Choices

**Single Fluid vs Two Fluid?**

- **Single Fluid**
  - Fertile (Th) and fissile (U233) in a one carrier salt
  - Processing to remove fission products the most complex

- **Two Fluid**
  - Thorium blanket salt, Fuel salt no fertile, only U233
  - Pa removal not necessary
  - Fission product removal appears simpler
  - Core design “was thought” to be complex
  - Need to verify barrier materials

- **One and One Half Fluid**
  - ORNL term for a mixed fluid surrounded by a thorium blanket salt. Pros and Cons of both
Design Choices

Pa Removal?

- Pa removal is expensive, controversial and unnecessary option in just about any MSR design
- Even for the Single Fluid 1970s MSBR would still breed without Pa removal
- Two Fluid designs have no need at all as the Pa is diluted in the blanket salt (sees lower n flux)
- ORNL 3996 Two Fluid looked at both ways
- Pa removal processing 100 tonnes of blanket salt TWICE a day gave B.R. 1.07
- Skipping this altogether, B.R. 1.05
- Can just dilute with more blanket salt if you want
Design Choices

**Harder or Softer Spectrum?**

- **Harder Spectrum** *(fast)*
  - Can skip graphite use and easier to breed
  - Takes far more fissile material to startup
  - Avoiding neutron leakage and reactor vessel damage a challenge
  - Typically only larger and high power *(GWe+)* enables breeding *(small is too leaky)*
  - Accidently criticality of spills an issue

- **Softer Spectrum**
  - Reactor control is easier
  - Much smaller fissile startup
  - Must remove fission products quickly to breed
  - Graphite used
Design Choices

Breeder vs Converter?

- **Breeder**
  - Requires processing of the salt to continuously remove fission products

- **Converter (i.e. burner)**
  - Needs annual fissile makeup
  - Can skip fuel processing
  - Less R&D needed
  - Core design greatly simplified
Design Choices

Denatured vs Pure Th-U233?

- **Pure Cycle (Th-U233)**
  - Better neutron budget than U235 or Pu239
  - Does though mean use of Highly Enriched Uranium
    - U232 is a deterrent but effects often overstated
    - Can make plant secure but will need regulator flexibility
  - Startup difficult, U233 non-existent but can use Pu
  - Two Fluid designs can transition from Low Enriched Uranium
  - For converter designs makeup fissile a challenge

- **Denatured (enough U238 present)**
  - Eases regulatory concerns, LEU used (with or without Th)
  - Modest drop in conversion ratio but ORNL TM 6413 showed denatured breeder possible in 1980
  - Converter versions will need slightly more annual fissile makeup but as simpler to acquire LEU
Design Choices

**U233 and the IAEA**

- IAEA regulations require a U232 fraction of 2.4% to be deemed “self protected”
- MSR designs 0.2% likely a maximum with a great deal of variation between studies
  - ORNL calculated only 0.002% U232 for the MSBR (20 ppm)
- There are ways to increase U232/U233 ratios
- Th230 can be found in Uranium mining waste
  - If extracted, helps lower radioactivity of tailings
  - Adding Th230 will significantly increase U232 production via Pa231 (yet to be published work of Bruce Hoglund)
- U233 makeup fuel from Fusion/Fission Hybrid
  - A 1000 MWe fusion reactor could produce 2000 kg of U233 per year with 5% U232 content
  - As a makeup fuel, the added U232 will burn off slower than the added U233
Thermal MSR Converter using Fusion produced U233 makeup (5% U232)

Very recent work (unpublished) by Ralph Moir fusion/fission expert retired from Lawrence Livermore Lab

If MSR Converter has a conversion ratio of less than 0.9 then its contained uranium will be “self protecting”
Design Choices

Fueled vs Cooled?

- Existing nuclear community may be more comfortable with “cooled” or FHR
  - 95% of any R&D applicable to “fueled”

- Compared to existing reactors, only safety appears to have a clear advantage
  - Economic advantage marginal
  - Resource utilization similar to LWR
  - Waste profile arguably inferior
Reexamining MSRs

- MSRs often thought of as the “thorium” reactor
- By mandate they were developed as breeders to compete with the Sodium Fast Breeder
- The belief at the time was Uranium resources were extremely limited, we now know better
- MSRs can be both “converters” or “breeders” but choices must come down to pragmatic facts, not ideology or imposed funding mandates
- However, no one can dispute the success of advancing “thorium” to the public

Come for the Thorium
Stay for the REACTOR!
Back to *Breeder vs Converter*

- Most researchers focus on pure breeders
- However, the R&D and operational costs of continuous salt process much higher than most assume
- A pure Th-U233 cycle also involves Highly Enriched Uranium. Many consider this a non-starter on proliferation grounds
- A “converter” has almost negligible fuel costs, assured resources, enhanced anti-proliferation features and overall is much simpler and less R&D
My Design Philosophy

- “A designer knows he has achieved perfection not when there is nothing left to add, but when there is nothing left to take away.”
  Antoine de Saint-Exupery

- Aim has been to simplify (or remove!) as much as possible
- Removing as much technological uncertainty and needed R&D crucial to obtain private sector support
Proposed Design Routes*
*All patent pending

- **Tube in Tube Two Fluid**
  - Solving ORNL’s “Plumbing Problem”

- **The “Nuclear Sparkplug”**
  - Helps solve neutron leakage and vessel damage issues of otherwise attractive Single Fluid harder spectrum designs

- **New and Improved DMSR Converter**
  - Many new changes focused on simplification and minimizing R&D
  - Will likely be main focus of efforts
  - Some of these advances may also improve salt “cooled” FHR approach
ORNL’s Two Fluid “Plumbing Problem”

- ORNL recognized the many advantages of Two Fluid design and was their main focus from the early 1960s to 1968
- Design was hundreds of interlaced graphite tubes separating fuel and blanket salts
ORNL’s Two Fluid “Plumbing Problem”
Why interlace the Two Fluids?

- As graphite changed dimensions, so did the ratio of fuel to blanket salt
- Blanket salt would have large positive void coefficient
- Abandoned in 1968 in favor of Single fluid
- ORNL would explain that they viewed interlacing unavoidable as trying a single central zone of fuel salt would be too small to produce enough power (~1 meter in diameter is a max, perhaps a few 10s of MWe)

**HOWEVER.....**
Tube within Tube Geometry

Side View of Reactor Core and Surrounding Blanket Salt

Core is Graphite + Fuel Salt or 100% Fuel Salt
Typical Diameter of 1 meter

Core becomes subcritical

Active Core

Blanket Salt

Blanket Salt

Hastelloy N or graphite to heat exchanger
Blanket Salt 60 cm thick

Expands power producing volume while maintaining the small inner core needed for a simple Two Fluid design
New Concepts Advantages

- Can use simpler Two fluid fuel processing without the “plumbing problems”
- No need for graphite moderation
- 100 to 200MWe possible per tube core
- Very strongly negative fuel salt coefficients
- Blanket will also have negative temp/void coefficient as it acts as a weak reflector
  - Make sure final outer vessel is neutron absorbing
- Simple, transportable cores
- Fissile inventory as low as 400 kg per GWe or even lower. Start up on LEU an option (for ANY Two Fluid design)
Potential Disadvantages

- Like any Two Fluid design, a barrier within neutron flux must be maintained
  - All Two Fluid designs run fissile salt at lower pressure (any leak will be blanket inwards)

- Core is not in minimally reactive configuration (no Two Fluid design ever is)
  - If graphite is used, run horizontally
  - Even worst case scenario would only appear to make an expensive mess
Critical Issue: Core-Blanket Barrier

- Viability of barrier materials in high neutron flux main R&D need
- Much recent work in the fusion field using $^{27}$LiF-BeF$_2$ salt as coolant
- Graphite, Molybdenum, SiC/SiC or simple carbon composites leading candidates
- Ease of “retubing” means even a limited lifetime still may be attractive
Nuclear Sparkplug

- Solves the problem of outer vessel damage, neutron leakage or need for a blanket or barrier for harder spectrum designs
- Replaceable component that makes the difference
- Large list of advantages and easily well above a C.R. of 0.9 in Converter Mode (no processing)
  - Well under 20 tonnes U per GWe year
- By adding modest continuous processing break even breeding with ease (as simple as U+Pu fluorination and salt discard)
- Like Tube and Tube though, moderate technological uncertainty and R&D needed
Sorry, no details for now…
If simplicity the goal? What design?

- Majority of my work has been improvements to final MSR work of ORNL in the late 1970s, the Denatured Molten Salt Reactor
- Growing number of advocates agree simplified converters is the right approach
- Quote from Xu Hongjie, head of Chinese program, ThEC 2012 question period:
  - “If you just want to use thorium as a fuel you don’t need C [conversion ratio] larger than 1, less than 1 is OK, maybe 0.8, you just assume consuming uranium with the thorium”
- It’s well known conversion ratio of the DMSR was 0.8 and would get more than half its energy through the thorium chain
DMSR Converter Reactors

- Oak Ridge`s 1000 MWe 30 Year Once Through Design (1980)
- Originally mandate to increase anti-proliferation
- 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)
- Similar fissile startup load to LWR (3.5 t/GWe)
- Better reactivity coefficients than MSBR
  - MSBR -0.9 pcm/K
  - Grenoble Recalculation of MSBR ~+1 pcm/K
  - DMSR -6.8 pcm/K
Denatured Molten Salt Reactors

- When salt finished, option to process and recycle
- Uranium simple and economical to remove, transuranics should also be recycled
- Have up to 30 years to acquire equipment
- Under 1 tonne TRUs in salt at shutdown
- Assuming typical 0.1% processing loss, less than 1 kg in 30 years! As good or better radiotoxicity as pure Th-233U cycle
- Reducing the Earth`s Radioactivity?
  - After 300 years, less radiotoxicity exists than before the reactor started (mainly from natural U-234)
  - No other reactor can make this claim
Ingestion Radiotoxicity PWR vs FBR* vs MSR*

*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*
Ingestion Radiotoxicity PWR vs FBR* vs MSR*

*Assuming 0.1% Loss During Processing

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

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**Radiotoxicity R(t) of actinide waste**

- **PWR U Ore**
- **DMSR U Ore**
- **MSBR Th Ore**

**FPs**  
Fission Products

**Turns waste management into 500 year job, not million year**
<table>
<thead>
<tr>
<th>Reactor</th>
<th>Lifetime Uranium (t)</th>
<th>Annual Uranium (t)</th>
<th>Annual Enrichment SWU</th>
<th>Annual Fuel Costs 100$/kg U $ millions</th>
<th>Annual Fuel Costs 2500$/kg U</th>
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<tbody>
<tr>
<td>LWR</td>
<td>5430</td>
<td>170</td>
<td>140,000</td>
<td>50</td>
<td>460</td>
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<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>0.6cent/kwh</td>
<td>7 cents/kwh</td>
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<tr>
<td>LWR with U-Pu Recycle</td>
<td>3460</td>
<td>106</td>
<td>87,500</td>
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<tr>
<td>DMSR</td>
<td>1525</td>
<td>30</td>
<td>35,100</td>
<td>8.6</td>
<td>80</td>
</tr>
<tr>
<td>Once Through</td>
<td></td>
<td></td>
<td></td>
<td>0.1cent/kwh</td>
<td>0.9 cents/kwh</td>
</tr>
<tr>
<td>DMSR with U Recycle</td>
<td>850</td>
<td>30</td>
<td>35,100</td>
<td>8.6</td>
<td>80</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>0.1cent/kwh</td>
<td>0.9 cents/kwh</td>
</tr>
<tr>
<td>DMSR 15 Year Batches</td>
<td>~500</td>
<td>~17</td>
<td>~20,100</td>
<td>4.9</td>
<td>46</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
<td>0.07cent/kwh</td>
<td>0.7 cents/kwh</td>
</tr>
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</table>

Based on 0.2% tails, 75% capacity factor, 30 year lifetime
LWR data from “A Guidebook to Nuclear Reactors” A. Nero 1979
Fuel cost assume 160$/SWU, 10$ million for LWR fuel fabrication
Above $1000/kg uranium resources likely unlimited
How does a DMSR do so well?

- Isn’t Heavy water better than graphite
- Far less parasitic losses
  - No internal structure
  - No burnable poisons
  - Less leakage
- LWR 22% parasitic losses (without FPs)
- CANDU 12%
- DMSR 5%
- Almost half of fission products and all important Xe135 leave to Off Gas system
- Fissile produced in situ is almost all burned in situ. LWRs and CANDU throw most out
Suggested Improvements

- Shorter batch cycles of the salt
- As long as U is recycled (TRUs can wait) large improvement in U needs
- 10 to 15 year batches likely 20 t U per GWe year and 24,000 SWU
- Current U mining and enrichment could support 2500 GWe of such DMSRs (current entire world’s electricity)
Suggested Improvements

- Higher Power Density
  - Much smaller cores
  - Smaller building and less startup fissile
  - Need to replace Graphite periodically

- Allow a lower conversion ratio???
  - Fuel cost so low and only cheap Uranium limited
  - Wise to examine simplifications that might slightly rise fuel costs from 0.1cents/kwh but save effort and more money elsewhere
Running without thorium has many interesting advantages

Neutron economy not as quite as good but still excellent uranium utilization

No Protactinium
  - Can run any power density or spectrum

Lower melting points

Simpler to re-enrich uranium (no U232) to recycle Uranium indefinitely
Suggested Improvements

- Alternate carrier salts
  - **NaF-BeF$_2$** low cost, low melting point
    - May allow stainless steel throughout loop
    - Less neutron loss than most assume
  - **NaF-RbF** low cost, no tritium production
    - Simplification of entire primary loop
  - **NaF** very low cost, higher melting point
    - Annual uranium only minor increase

- Smaller Power Outputs
  - Slightly higher U consumption (leakier cores)
  - Far less R&D and investment to demonstrate
  - Perhaps new innovation already exists???
ORNL FHR Focus on Solid Block Fuel Forms

- Complexity of online pebble handling equipment removed
- Dynamic pebble packing fractions may still be of some concern for regulators
- Far lower pumping power than pebble beds
- Focus on 1500 MWe AHTR design and small 50 MWe SmAHTR modular unit
- One drawback is fuel must contain considerable burnable poisons to give a practical time between refueling
- This leads to disappointing Uranium utilization
  - Higher starting fissile load
  - Upwards of double uranium needs of LWR
SmAHTR 50 MWe

Many attractive innovations developed for SmAHTR, also attractive for DMSR
Thanks ORNL
From “cooled” back to “fueled”?
Thanks ORNL

From “cooled” back to “fueled”?

- Integration of IHX within core and keeping vessel top away from salt and neutron flux a great idea
- Basic idea is take this and replace TRISCO core with simple graphite and put fuel back into the salt
- Short shutdowns to open vessel and replace graphite and/or heat exchangers every 4 years
- Easily go to higher power density and 100 to 200 MWe. Units can be combined for even larger plants
- Obviously want to reduce out of core salt volume
Design Effort Summary

- Believe two new and unique patent pending designs may be *best* MSR options
- “*best* is often the enemy of *good enough*”
- Focus has shifted to making the DMSR approach as practical and as simple to develop as possible with many exciting results
- I look to borrow from FHR “salt cooled” work (SmAHTR) but looking also to give back
  - Patent pending DMSR innovations also applicable to FHR
  - A few innovations specifically for FHR
Where To From Here?

- Worldwide interest in Molten Salt Reactors continues to grow
- Safety case, improved waste profile and resource stability obvious selling points to the public
- Economic case has potential to win over governments and corporations
- For the private energy sector, long horizons always a tough sell
- BUT, since industrial heat also our commodity, entire industries may realize they can’t afford NOT to get involved
- Ex. Steam for Canadian Oil Sands Extraction
  - 15 to 25 year development horizons normal there
Canadian Focal Point?

- Strong Nuclear Community going idle as “advanced” CANDU work halted
- University sector and Chalk River Nuclear Laboratories very interested
- New major Canadian Corporate Player
- CNSC far less “inertia” than NRC and actively promoting Canada for Small Reactor Development
- Oil Sands developers could fund entire DMSR program from pocket change
  - When oils gone, still a piece of the energy pie
Introducing…

Terrestrial Energy Inc.

- Just founded by core group with diverse financial and entrepreneurial expertise including Oil Sands insiders all drawn to MSR’s potential
- My job, with the help of gathered talent, to further refine, evaluate and consolidate IP for the one or two most cost and R&D efficient MSR and/or FHR designs possible
- Their job, to attract the modest investment and industrial partners needed to get to the conceptual blueprint stage
- Followed by the more challenging stage of funding a demonstration reactor
- As IP is disclosed, I hope to make their job easier
Thank you for your attention

Keep in mind
Thorium Energy Alliance Conference
May 2013 Chicago
EXTRA SLIDES…
Two Region Homogeneous Reactor
Projected breeding ratios assume thicker blanket and alternate barrier. From ORNL 2751, 1958

<table>
<thead>
<tr>
<th>Core Diameter</th>
<th>3 feet</th>
<th>4 feet</th>
<th>4 feet</th>
<th>8 feet</th>
</tr>
</thead>
<tbody>
<tr>
<td>ThF$_4$ in fuel salt mole %</td>
<td>0</td>
<td>0</td>
<td>0.25</td>
<td>7</td>
</tr>
<tr>
<td>$^{233}$U in fuel salt mole %</td>
<td>0.592%</td>
<td>0.158%</td>
<td>0.233%</td>
<td>0.603%</td>
</tr>
<tr>
<td>Salt Losses</td>
<td>0.087</td>
<td>0.129</td>
<td>0.106</td>
<td>0.087</td>
</tr>
<tr>
<td>Core Vessel</td>
<td>0.090</td>
<td>0.140</td>
<td>0.109</td>
<td>0.025</td>
</tr>
<tr>
<td>Leakage</td>
<td>0.048</td>
<td>0.031</td>
<td>0.031</td>
<td>0.009</td>
</tr>
<tr>
<td>Neutron Yield</td>
<td>2.193</td>
<td>2.185</td>
<td>2.175</td>
<td>2.20</td>
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<tr>
<td>Breeding ratio (Clean Core)</td>
<td>0.972</td>
<td>0.856</td>
<td>0.929</td>
<td>1.078</td>
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<tr>
<td>Projected B.R. (thinner wall)</td>
<td>1.055</td>
<td>0.977</td>
<td>1.004</td>
<td>1.091</td>
</tr>
<tr>
<td>Projected B.R. (carbon wall)</td>
<td>1.105</td>
<td>1.054</td>
<td>1.066</td>
<td>1.112</td>
</tr>
</tbody>
</table>
Estimating C.R. for shorter batch cycles (15 years or less)

Eta 1.99
To start

Eta 2.10 (more U233)
At 15 years
C.R. back to 0.8

New average C.R. can attain 0.85 to 0.9 for 10 to 15 year batches.
About 1000kg fission per Gwe year so as low as 100kg shortfall = 22.8 t at 0.2% tails or 17.7 t at 0.05% tails

Ave Eta values from next slide

Fig. 6. Conversion ratio vs time.

Eta - C.R. – 1.0 = 0.19
(early parasitic losses) If restarted with eta 2.1 at 15 years expect early C.R. = 2.1 - 1 - 0.19 = 0.91
Table 10. Nuclide concentrations and neutron utilization after 15 years of DMSR operation

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Concentration ($\times 10^{24}$)</th>
<th>Neutron absorption$^b$</th>
<th>Fission fraction</th>
<th>$\nu \sigma_f / \sigma_a$</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{232}$Th</td>
<td>2.561</td>
<td>0.2561</td>
<td>0.0017</td>
<td>0.0070</td>
</tr>
<tr>
<td>$^{233}$Pa</td>
<td>1.13</td>
<td>0.0018</td>
<td>0.0000</td>
<td>0.0033</td>
</tr>
<tr>
<td>$^{233}$U</td>
<td>49.0</td>
<td>0.2483</td>
<td>0.5480</td>
<td>2.2427</td>
</tr>
<tr>
<td>$^{234}$U</td>
<td>9.21</td>
<td>0.0120</td>
<td>0.0002</td>
<td>0.0143</td>
</tr>
<tr>
<td>$^{235}$U</td>
<td>25.1</td>
<td>0.1161</td>
<td>0.2272</td>
<td>1.9894</td>
</tr>
<tr>
<td>$^{236}$U</td>
<td>16.2</td>
<td>0.0075</td>
<td>0.0001</td>
<td>0.0168</td>
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<tr>
<td>$^{237}$Np</td>
<td>1.83</td>
<td>0.0047</td>
<td>0.0000</td>
<td>0.0102</td>
</tr>
<tr>
<td>$^{238}$U</td>
<td>476</td>
<td>0.0901</td>
<td>0.0017</td>
<td>0.0194</td>
</tr>
<tr>
<td>$^{239}$Pu</td>
<td>4.34</td>
<td>0.0896</td>
<td>0.1578</td>
<td>1.7905</td>
</tr>
<tr>
<td>$^{240}$Pu</td>
<td>2.46</td>
<td>0.0324</td>
<td>0.0001</td>
<td>0.0032</td>
</tr>
<tr>
<td>$^{241}$Pu</td>
<td>1.84</td>
<td>0.0293</td>
<td>0.0628</td>
<td>2.1754</td>
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<tr>
<td>$^{242}$Pu</td>
<td>2.38</td>
<td>0.0039</td>
<td>0.0001</td>
<td>0.0136</td>
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<tr>
<td>Transplutonium$^c$</td>
<td></td>
<td>0.0014</td>
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<tr>
<td>$^{238}$Pu</td>
<td>0.882</td>
<td>0.0024</td>
<td>0.0003</td>
<td>0.1245</td>
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<tr>
<td>Total actinides</td>
<td></td>
<td>0.8956</td>
<td>1.0000</td>
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<tr>
<td>Fluorine</td>
<td>48,000</td>
<td>0.0079</td>
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<tr>
<td>Lithium</td>
<td>24,500</td>
<td>0.0062</td>
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<tr>
<td>Beryllium</td>
<td>5,470</td>
<td>0.0012</td>
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<td></td>
<td></td>
<td></td>
<td>0.9109</td>
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<tr>
<td>Graphite</td>
<td>92,270</td>
<td>0.0172</td>
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<tr>
<td>Fission products</td>
<td></td>
<td>0.0563</td>
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<tr>
<td>Total</td>
<td></td>
<td>0.9844</td>
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</tr>
</tbody>
</table>

$^a$ Nuclei per cubic meter of salt or moderator.

$^b$ Absorption per neutron born; leakage is 0.0156.

$^c$ Includes $^{240}$Pu, $^{241}$Pu, and $^{242}$Pu produced from a decay of $^{244}$Cm.
74%LiF-16.5% BeF2-9.5%(U,Th)F4 \textit{versus} 53%NaF-20%RbF-27%(U,Th)UF4 mp 500 C

- New salt has 77% more heavy atom density
- Can thus run at lower relative salt fraction in core (same Carbon to Fissile Ratio)
- Na for thermal neutrons is only 6 times the absorption cross section of 99.995% $^7$LiF
- In DMSR, more losses to fluorine than Li
  - Will have 56% less fluorine, 65% less Na+Rb compared to Li (and no Be)
  - Works out to roughly same neutron loss (estimate only, ignores higher resonances bands for Na and Rb)
LWR Fuel Cycle Costs

All assume 100$/kg U and 150$/kg SWU

- **Light Water Reactor (per Gwe)**
- 20M$ Uranium (200 Tonnes, 100$/kg)
- 20M$ Enrichment
- 10M$ Fuel Fabrication
- Annual Fuel Cost **0.6 cents/kwh**
- **But** must pay off initial fuel load
  - 3 to 5 Tonnes U235 + Fabrication
  - ~200M$ = 0.26 c/kwh (10% Discount Rate)

- Total Fuel Cycle ~ **0.86 cents/kwh**
Fast Breeder Fuel Cycle Costs

- Sodium Fast Breeder
- Capex of reprocessing equipment???
- Fabrication costs?
- Looking *only* at the initial load of fissile*
- Need ~ 18 Tonnes Reactor Grade Pu (12 T fissile)
  - 100$/gram = 1.8 B$ = \textbf{2.3 c/kwh}
- Or start on ~ 20 T U235 at 50$/g = 1.25 c/kwh
  - 20 years worth of U235 for a LWR
  - Over a hundred years worth for a DMSR
- This does ignore Pu production credit but processing costs must be factored in
- IFR or TWR are about half traditional FBR startup

* Yes, there are current stockpiles of separated Pu but limited supply and paid by tax payers.
MSR Fuel Cycle Costs

Estimate based on ORNL 4541 and 4812 (1972) and 7.5 times “nuclear” inflation
Old 75% capacity factor but new 10% discount rate (ORNL was 13.5%)

- Single Fluid Graphite Breeder (MSBR 1970s)
  - 1500 kg/GWe starting fissile load
  - 150 M$ fissile 100$/g U233 or Pu = 0.23 c/kwh
  - Annual and startup Thorium = negligible
  - Starting and makeup salt = 0.07 cents/kwh
    - Enriched Lithium costs an unknown factor but likely low impact
- 10 day processing cycle
- Processing Plant Cap + Op
  - ORNL 4541, 100M$  0.16 cents/kwh
  - ORNL 4812, 260M$  0.4 c/kwh
- Sounds high but still only 5$/kg of salt processed
  - PUREX is 1000$/kg

- Total Fuel Cycle = 0.46 to 0.7 cents/kwh
  - Large Processing cost uncertainty
MSR Fuel Cycle Costs

Estimate based on ORNL 4541 and 4812 and 7.5 times “nuclear” inflation
Old 75% capacity factor but new 10% discount rate (ORNL was 13.5%)

- Fast Spectrum (MSFR 2005 to present)
  - 5.5 T/GWe, 6 month processing time
  - 550 M$ fissile 100$/g U233 or Pu= 0.85 c/kwh
  - Less starting and makeup salt = 0.03 cents/kwh
  - Processing Plant Cap + Op
    - Much lower rate (1/18th MSBR) but economy of scale lost?
    - Also need process blanket salt
    - ~0.1 cents/kwh? Perhaps much higher?

- Total Fuel Cycle ~ 1 cent/kwh
  - Again large cost uncertainty
MSR Fuel Cycle Costs

Estimate based on ORNL 4541 and 4812 and 7.5 times “nuclear” inflation
Old 75% capacity factor but new 10% discount rate (ORNL was 13.5%)

- Single Fluid Graphite Converter (DMSR 1980)
- Runs off Thorium plus Low Enriched Uranium
- 3450 kg/GWe starting fissile, **No processing**
- 175 M$ fissile 50$/g U235 = 0.26 c/kwh
- Annual and startup Thorium = negligible
- More carrier salt = 0.14 cents/kwh
  - Converter able to use inexpensive alternate salts
- Average Conversion Ratio 0.8 over 30 years
  - ~150 kg/year U235 (in LEU)
  - 50$/g = 7.8M$/year = **0.12 cents/kwh**

- **Total Fuel Cycle = 0.52 cents/kwh**
  - Very little uncertainty
  - 9 T of Pu (IFR startup) would start and run for over 30 years
  - Great potential for improvement
MSR Fuel Cycle Costs

- If a thorium based MSR does not break even, needs makeup of Pu or U233
  - Uncertainty of supply and high cost
- If a Uranium or U+Th based MSR does not break even, simply makeup with LEU
- A DMSR can be modified to have even better Conversion Ratio still without processing
  - C.R. = 0.9, ~20 tonnes U/GWe-year, 0.06 c/kwh
- Or, a DMSR can be further simplified, low cost salts, lower cost graphite, less fissile startup etc.
  - C.R. = 0.7 ~60 tonnes U/GWe-year, 0.18 c/kwh

- More on the DMSR later....
What factors differentiate between various Molten Salt designs?

- R&D required and level of technological uncertainty
- Amount and type of startup fissile load and thus deployability
- Whether fission product removal is used and if so, its degree of difficulty
- Reactivity coefficients
- Degree of Proliferation Resistance
Comparing Designs

- Mid 1960s Two Fluid
- Late 1960s Single Fluid
- DMSR Breeder and/or Burner
- Japanese FUJI
- European 1 ½ Fluid MSFR
- New Tube within Tube design
- DMSR Revisited
Mid 60s ORNL Two Fluid MSBR
Mid 60s ORNL Two Fluid MSBR

Advantages

- Much easier removal of fission products as no thorium in fissile salt
- Only graphite in strong neutron flux
- Strong negative temp coefficient for fissile salt
- Very low fissile inventory
  - 700 kg per GWe
Disadvantages

- Core plumbing a huge challenge as graphite shrinks then swells
- A single tube failure means entire core replaced
- Strongly Positive temp coefficient for blanket salt
Single Fluid Graphite MSBR (70s)

Advantages

- Relatively simple core
- No structural material or barriers needed within strong neutron flux
- Modest starting inventory (1.5 t/GWe)
- High thermal inertia (slow to change temperature)
- As with any practical design, negative temperature coefficient (at least initially)
Single Fluid Graphite MSBR (70s)  

**Disadvantages**

- Complex and rapid fission product removal with much R&D needed
- The longer term reactivity coefficient (10s of seconds after any power surge) *may* be slightly positive
- To start, needs hard to obtain U233 or LWR transuranics which are of limited availability for large fleet deployment
Late 1970s DMSR *Breeder*

- ORNL late 70s version starts on Low Enriched Uranium and Thorium
- Larger, lower power density to get full 30 year life from graphite
- Added benefit of much better reactivity coefficients than MSBR
- However, even more complex fission product removal to barely break even
Late 1970s DMSR Converter
30 Year Once Through Design

- No fission product removal
- Needs small amounts of LEU annually
- More on this later...
The FUJI Approach

- Poorly funded but dedicated work has continued in Japan led by Dr. Kazuo Furukawa for many decades*

- Earliest FUJI designs
  - No graphite replacement
  - Batch salt treatment every several years
  - However, runs purely on Th-U233 cycle with external sources of U233 or TRUs needed

- New Harder Spectrum FUJI
  - More recent work in which a much higher fissile load and harder spectrum employed
  - Projected break even by batch salt treatment every few years

* Prof Furukawa passed away Dec 14th 2011, our condolences to his family, he will be sorely missed
Traditional FUJI

- Advantages vs DMSR
  - Less annual fissile needs than the DMSR (better conversion ratio)
  - Does not require any natural uranium or enrichment facilities

- Disadvantages vs DMSR
  - Start up and makeup fissile very difficult to obtain
  - U233 by accelerator expensive and needs separate R&D funds
  - Spent Fuel Pu expensive to remove and limited supply
  - Proliferation resistance of concern, especially the need for long term shipments of U233
Harder Spectrum FUJI

- Requires very large fissile load at startup of 7.8 t U233 per GWe
  - Electricity bill alone to produce that much U233 from accelerators almost 2 billion$

- Starting on TRUs likely unfeasible due to PuF$_3$ solubility limits
1 ½ Fluid MSFR (was TMSR)

Design has a thorium blanket but only radial, not axially (which would be very difficult)
1 ½ Fluid MSFR (was TMSR)

Advantages

- Much lower daily processing rate than the MSBR (but more complex)
- No graphite to replace
- Very good fuel reactivity coefficients
- Compact, fairly simple core
- Very high breeding ratio possible (upwards of 1.12 vs 1.06 of MSBR)
1 ½ Fluid MSFR (was TMSR) 

Disadvantages

- Large fissile inventory, 5 to 8 t/GWe
- Now calls for higher temp of 800 °C (assumed to assure solubility of PuF$_3$)
- Major materials R&D needed for blanket zone and axial reflectors and how to replace them (higher temp and strong neutron flux)
- Very small thermal inertia (15 m$^3$ salt)
Two Proposed Design Routes

- A return to the Two Fluid design
  - Solving the Plumbing Problem

- The DMSR Converter
  - Far fewer technical, economic and political obstacles
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
Advantages of all Molten Salt Reactors

Low Capital Costs

- Molten salts are superior coolants so heat exchangers and pumps are smaller and easy to fabricate
- This has a trickle down effect on building design, construction schedules and ease of factory fabrication
- Much higher thermal efficiency than LWR or FBR using Steam or Gas (He, CO2, N2)
- Fuel cycle costs extremely low
- No need for elaborate “defence in depth” or massive internal structures for steam containment and water reserves
Comparing Heat Exchange Equipment

**MSBR vs PWR vs Sodium FBR**

- **MSR 1/3 the total volume of PWR**
- **MSR 1/9 the total volume of FBR**
Advantages of all Molten Salt Reactors

Safety

- No pressure vessel
- No chemical driving forces (steam build up or explosions, hydrogen production etc)
- Almost no volatile fission products in salt
  - They are passively and continuously removed
  - Both Cesium and Iodine stable within the salt
- No excess reactivity needed
  - Even control rods are optional
- Very stable with instantly acting negative temperature reactivity coefficients
- Passive Decay Heat removal
All radiation within a sealed “Hot Cell”

Red shows Secondary Containment
Outer building is third level
Our message to Regulators?

- Nothing inside the Hot Cell can force its way out
- Any needed penetrations securely sealable
- Outer building protects from outside threats
- Regulate as “Hot Cell” not unlike a PUREX Plant?
Off Gas Both a Benefit and Challenge

- Dealing with fission products gasses and/or tritium a major challenge
- Many FPs have Xe or Kr precursors
  - Over 40% of FPs leave core
  - Large fraction of Cesium, Strontium and Iodine end up in Off Gas System
- ORNL work for 1000MWe plant
  - 2 hrs in drain tank (all Cs137) ~20MW
  - Then 47 hr delay charcoal beds ~2MW
  - 90 day long term beds ~0.25MW
  - 23 m3 of Kr+Xe a year in 8 gas bottles
Advantages of all Molten Salt Reactors

**Long Lived Waste**

- Fission products almost all benign after a few hundred years
- The transuranics (Np, Pu, Am, Cm) are the real issue and reason for “Yucca Moutains”
- All designs produce less TRUs and can be kept in or recycled back into the reactor to fission off
Radiotoxicity PWR vs FBR* vs MSR*

*Assuming 0.1% Loss During Processing

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

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**Turns waste management into 500 year job, not million year**
Advantages of all Molten Salt Reactors

Resource Sustainability

- Once started breeder designs only require minor amounts of thorium (about 1-10 tonne per GWe year)
  - 30 k$ of thorium = 500 M$ electricity
  - Must include processing costs though
- Converter designs are far simpler and only require modest amounts of uranium
  - Typically 35 tonnes U per GWe-year versus 200 tonnes for LWRs
  - Annual Fuel cycle cost $0.1 cents/kwh
For Better or Worse…
Quick Primer

- Molten Salt Reactors are fluid fueled with fluoride salts of U, Th and/or Pu dissolved in a carrier salt
- Salts flow in and out of a critical core and give heat to a secondary clean salt
- The core is typically a simple array of graphite but many different designs are possible
- Fission gasses are continuously drawn off and handled away from the core
- All designs excel in Safety, Cost, Long Lived Wastes and Resource Utilization
A Strange Beginning
An Aircraft Reactor?
A Brief History of Molten Salt Reactors

- Earliest efforts in support Aircraft Reactor Program in 1950s
  - Large knowledge base developed
  - Test reactor operates at 860 °C
- Major reactor development program at Oak Ridge National Labs late 1950s until mid 1970s
  - Major focus of ORNL
  - In competition with Sodium Fast Breeder
A Brief History of Molten Salt Reactors

- Very successful 8MWth MSRE test reactor from 1965 to 1969
- Design goes through several phases up to the Single Fluid MSBR (1968)
- In 1973, very controversial decision made to cancel program
- Limited work continued at ORNL until early 1980s, highlighted by the Denatured Molten Salt Reactor
A Brief History of Molten Salt Reactors

- Like all reactor programs, very little done in 1980s and 90s.
- Major boost in 2002 with MSRs chosen as one of six Gen IV designs.
- Strong programs in France, Russia, Czech Republic and especially China but near zero funding elsewhere.
- China reported to have 500 M$ MSR program with 400 scientist and engineers.
Uranium is not the enemy…

- Only “cheap” uranium is in limited supply
  - 500$/kg assures virtually unlimited supply
  - Still only 0.2 cents/kwh for “Burner” DMSR
- A few million tonnes U ore per year (51 kt U at world ave 3% ore grade)
- Compared to a few Gt (billion tonnes) iron and copper ore and 7 Gt of coal
- If uranium is used in DMSR designs, 100% of world’s electricity (2500 GWe) without increasing current mining
- Even if we needed to go to very low grade ore (0.03%) still only 200 Mt ore
I told you I’d bring this up Jess…

From Dr. Jess Gehin’s excellent presentation

There are several new technologies that must be introduced for deployment of Commercial MSRs

- >5 wt%U-235 LEU fuel (needed for startup)  
  - No
- U-233 as the fissile material  
  - No
- The use of thorium  
  - No
- Salt processing system (possibly online during operation)  
  - No
- Development and deployment of a new Li-7 enrichment capability  
  - No
- High temperature operation (potentially with a Brayton power conversion system)  
  - Not Really
- Off-gas handling and storage system  
  - Maybe
Proposed Pebble Bed DMSR Converter
Pebble Bed FHR (MIT, UCB, Wisconsin)

900 MWth FHR

400 MWth Gas Cooled
Pebble Bed FHR

- Pebbles can be cycled out so excess fissile and burnable poison not really needed

- Modest Uranium savings over LWRs (roughly CANDU levels)

- Newest version has Pebble stratified within core, varied by burn up
Testing Pebble Flow
UC Berkeley

Also uses complex arrangement of Axial and Radial flow to lower pumping power
The modular PB-AHTR is a compact pool-type reactor with passive decay heat removal.
Extremely High Proliferation Resistance

- Plant does not process the fuel salt
- Uranium always denatured, at no stage is it weapons usable
- Any Pu present is of very low quality, very dilute in highly radioactive salt and very hard to remove
  - About 3 times the spontaneous fission rate of LWR Pu and 5 times the heat rate (72.5 W/kg)
- No way to quickly cycle in and out fertile to produce fissile
What factors differentiate between various Molten Salt designs?

- R&D required and level of technological uncertainty
- Amount and type of startup fissile load and thus deployability
- Whether fission product removal is used and if so, its degree of difficulty
- Reactivity coefficients
- Degree of Proliferation Resistance
The World Needs Nuclear

- LWRs and HWRs mature technology but little area for improvements and widespread adoption unlikely
- Supercritical Water
  - Extremely challenging material science, still many years off
- Gas Cooled Prismatic or Pebble Beds
  - Good safety case, economics marginal
  - Must co-develop fuel fabrication and Brayton turbines
- Fast breeders
  - Decades and billions later, still unproven economics
- Small Modular LWR or FBRs
  - Fine for niche markets, unlikely a base load competitor
- Molten Salt Reactors have the potential to be true game changers
More on Molten Salt “Cooled” Reactors FHRs

- Basic concept is salt coolants are far superior to He or CO₂
  - TRISO fuel elements
  - Ambient Pressure
  - Can go to large total power and still have passive decay heat removal by natural convention of the salts
- Only “Flibe” 2Li²F-BeF₂ gives desired negative void coefficient
- Many involved would also favor true “fueled” MSRs but feel “cooled” is a more immediate or fundable step