National Status of LFR Development: USA

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Seminar: Activities for Lead-cooled Fast Reactors (LFR) in Generation IV International Forum (GIF)

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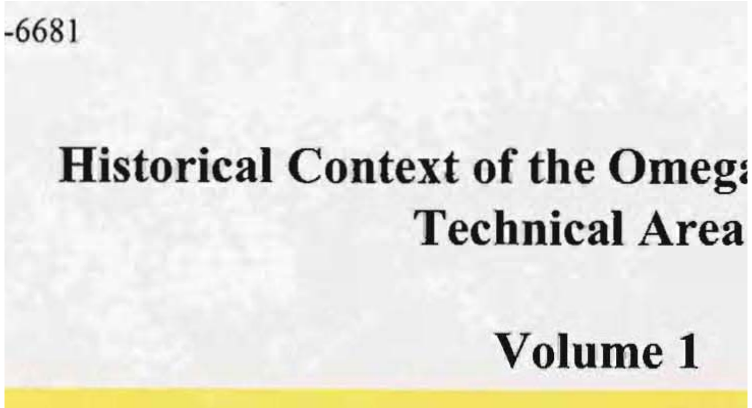
Outline of comments

- Historical backdrop: LMFR’s
- LFR Activities: current status
- Closing comments

US Fast Reactors: A Rich Historical Context

- **Clementine:** the first fast reactor, built by LANL in 1946. Pu-fueled, mercury-cooled, 25Wt power.
- **EBR-1:** the first reactor to generate electrical power, built in 1949-51 at Idaho Falls. NaK-cooled, 1.4 MWt, decommissioned in 1964. First breeder reactor.
- **Fermi-1:** 94MWe FBR prototype, built near Detroit in 1957, operated until 1972. Sodium cooled.
- **LAMPRE:** 1 MWt FR based on molten Pu fuel, sodium cooling, and reflector control, built at LANL in the 1957-61 time frame.
- **EBR-II:** sodium cooled 62 MWt reactor, built in Idaho as the IFR prototype in 1965; operated for 30 years.
- **SEFOR:** Experimental test reactor operated from 1969 to 1972 in Arkansas. MOX-fueled, Na-cooled, 20 MWt.
- **FFTF:** Built in 1978 in Washington State, 400 MWt Na-cooled test reactor.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Historical Context of the Omeg
Technical Area

Volume 1

Clementine
1946

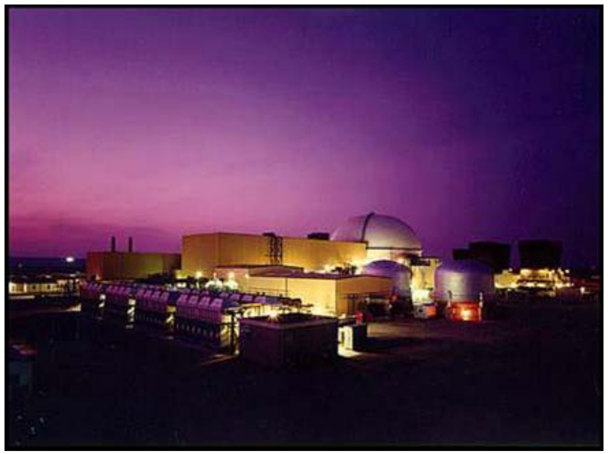
EBR-1
1951

Fermi-1
1957

EBR-2
1965

SEFOR
1972

FFTF
1978



Some chemical and thermal characteristics of liquid metal coolants

Coolant	Melting Point (° C)	Boiling Point (° C)	Chemical Reactivity (w/Air and Water)
Lead-Bismuth (Pb-Bi, LBE)	125	1670	Inert
Lead (Pb)	327	1737	Inert
Sodium (Na)	98	883	Highly reactive

Lead and LBE Coolants Provide Promising Overall Characteristics while Sodium Technology is More Highly Developed

LFR Compliance with Generation IV Goals

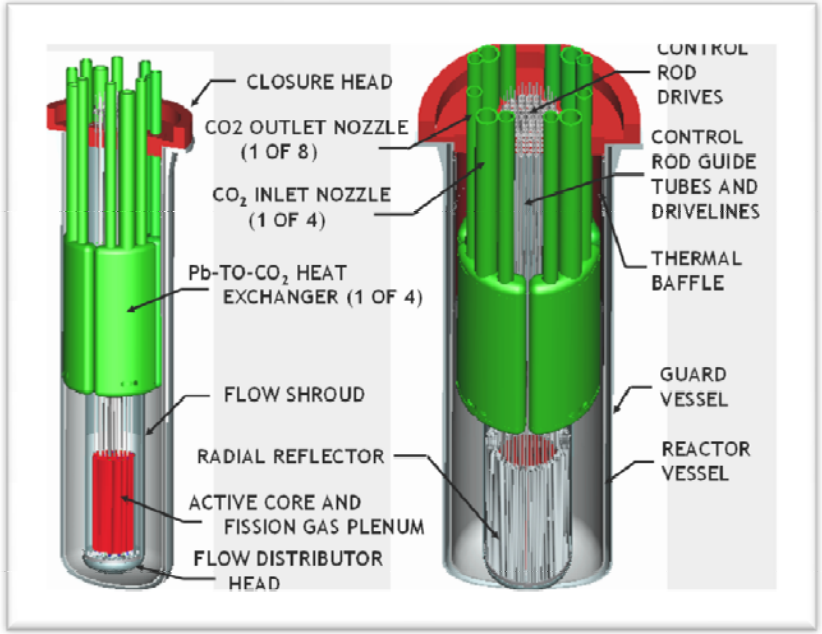
Goal Areas	*Goals achievable* via *intrinsic coolant properties plus Engineering*
Sustainability	▪ *Breeding gain close to 0* ▪ *Transmutation of MA*
Economics	▪ *Simplicity* ▪ *Compactness*
Safety and Reliability	▪ *Primary system at atmospheric pressure* ▪ *No risk of re-criticality in case of core melt (to be confirmed by severe accident analysis)* ▪ *Chemical inertness/high margin to boiling*
Proliferation Resistance and Physical Protection	▪ *Use of fuel containing MA* ▪ *Use of non-reactive coolant* ▪ *Sealed core and/or long refueling cycle*

LFR Activities in the US

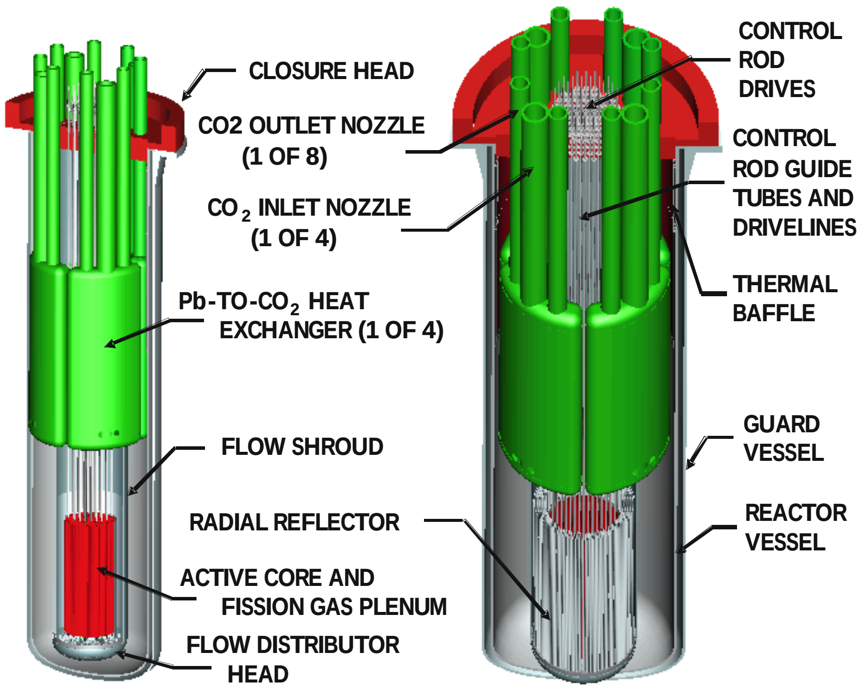
LFR activities in the US have been relatively limited in the past few years

Continuing (though mainly small) efforts include:

- Work at US national laboratories
- University efforts
- Some US industrial efforts
- SSTAR as a GIF reference concept for a small, transportable LFR



The Small Secure Transportable Autonomous Reactor (SSTAR)



SSTAR is a small natural circulation fast reactor of 20 MWe/45 MWt, that can be scaled up to 180 MWe/400 MWt.

The compact active core is removed by the supplier as a single cassette and replaced by a fresh core.

Key technical attributes include the use of lead (Pb) as coolant and a long-life sealed core in a small, modular system.

SSTAR Reactor Core Parameters

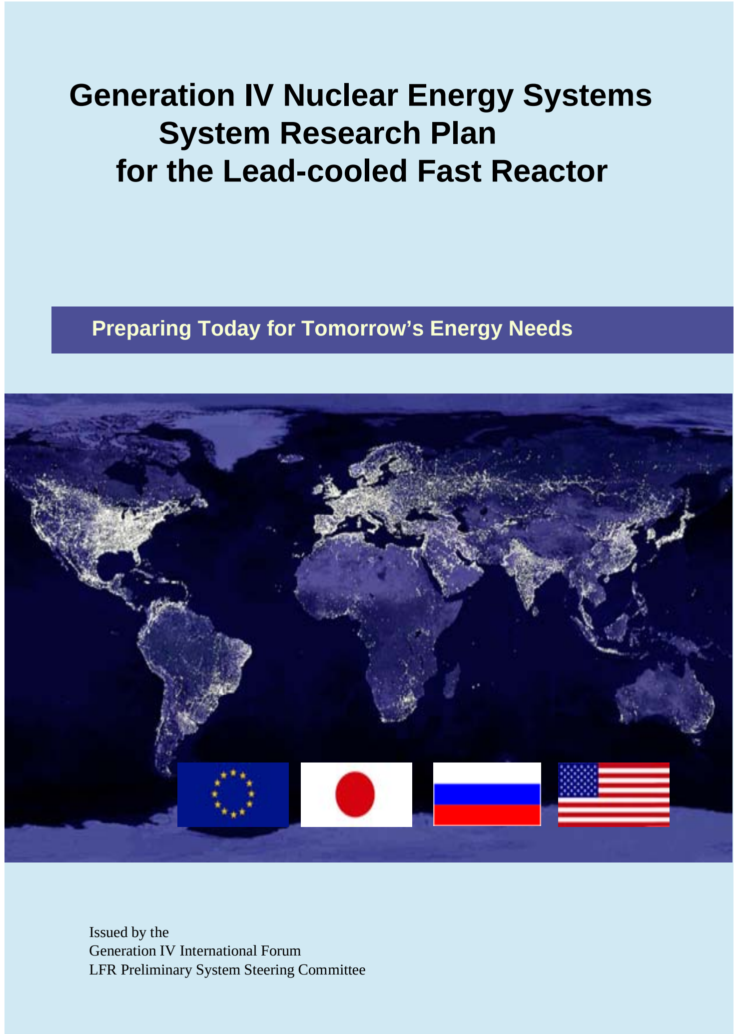
Coolant	Lead
Fuel	Transuranic Nitride, Enriched in N_{15}
Enrichment, %	5 Radial Zones, TRU/HM 1.7/3.5/ 17.2/19.0/20.7
Core Lifetime, years	15-30
Core Inlet/Outlet Temperature, ºC	420/567
Coolant circulation	Natural convection
Average (Peak) Discharge Burnup, MWd/Kg HM	81(131)

Peak Fuel Temperature, ºC	841
Peak Cladding Temperature, ºC	650
Fuel Pin Diameter, Cm	2.50
Fuel/Coolant Volume Fractions	0.45/0.35
Active Core Dimensions, Height/Diameter, m	0.976/1.22
Power conversion	$S\text{-}CO_2$ Brayton cycle

Recent and Continuing LFR Activities in the US

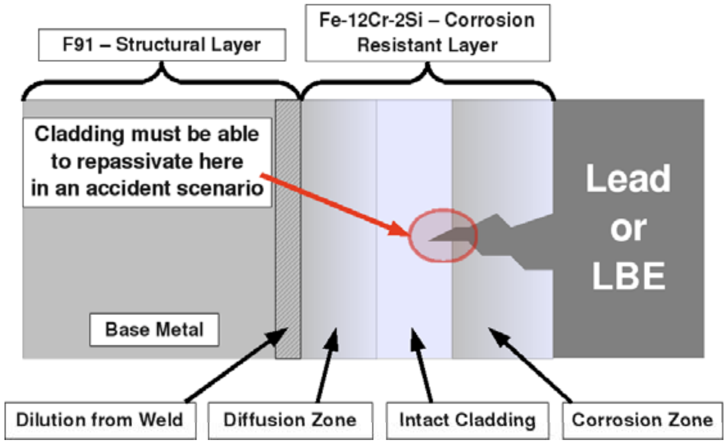
Efforts at US national labs

- LLNL support to DOE's Advanced Reactor program
- ANL completion of work on the 'SUPERSTAR' concept, an extension of the SSTAR concept
- LANL work with MIT and with UC Berkeley on material testing and performance.

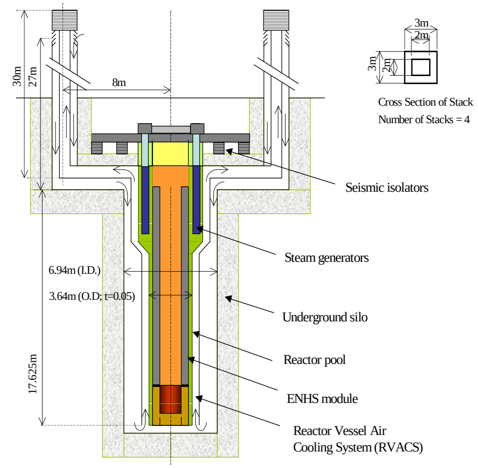


Continuing LFR Activities in the US

- University efforts
  - MIT work on Functionally Graded Composite materials
  - UC Berkeley material science and LFR design work
  - UNLV operation of an LBE loop



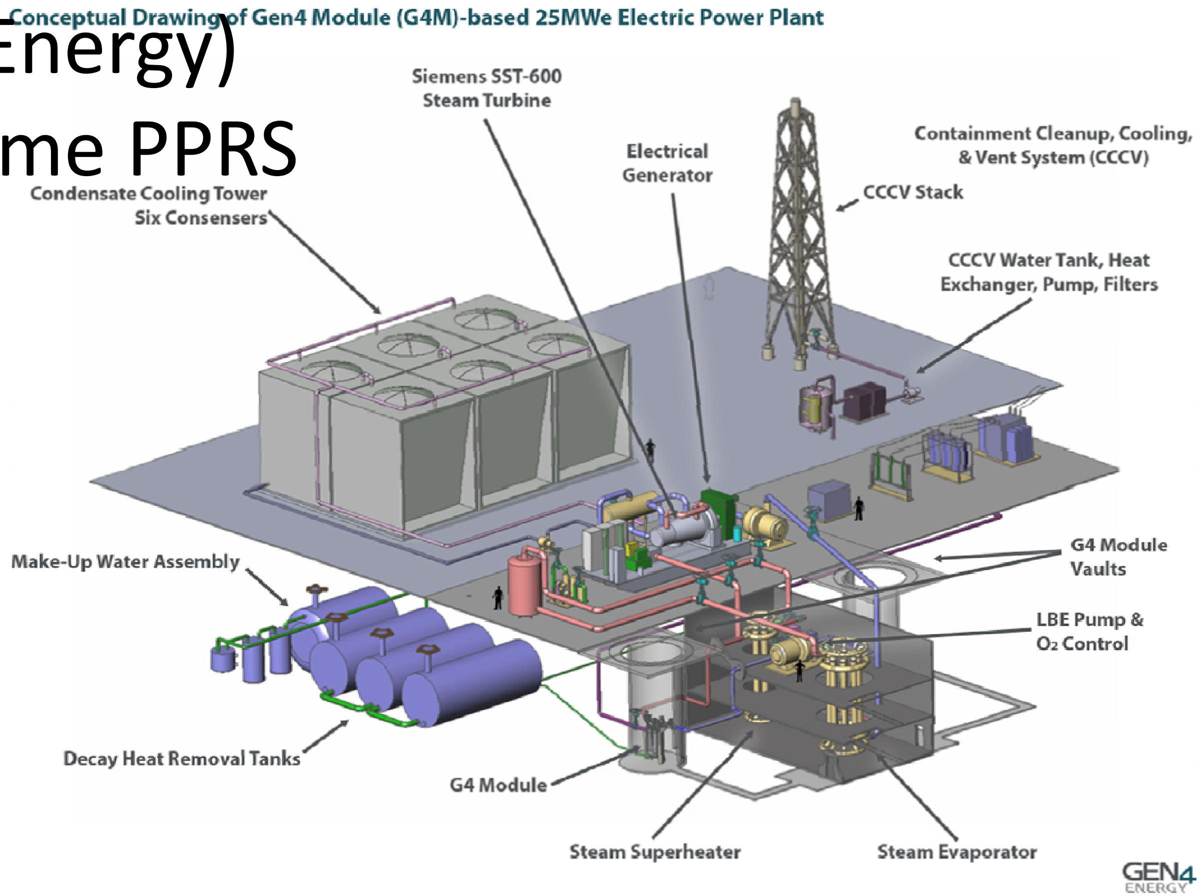
Schematic vertical cut through the ENHS reactor



Continuing LFR Activities in the US

Some limited industrial efforts in the US

- Hyperion Power Group (HPG) (now Gen-4 Energy)
- lakeChime PPRS



Gen-IV International Forum (GIF): Current Status with regard to LFR

- Preliminary System Steering Committee (pSSC) was formed in 2005
  - Members included EU, US, Japan and Korea
  - Prepared initial draft LFR System Research Plan (LFR-SRP)
  - Systems included a large central station design (ELSY) and a small modular (SMR) system (SSTAR)
- In 2010, an MOU was signed between EU and Japan causing a reformulation of the pSSC
- In 2011, the Russian Federation added its signature to the MOU
- In April, 2012, the reformulated pSSC met in Pisa and begun the process of revising the LFR-SRP
- The new pSSC envisions various updates to the central station and SMR thrusts while adding a mid-size LFR (e.g., the BREST-300) as a new thrust in the SRP

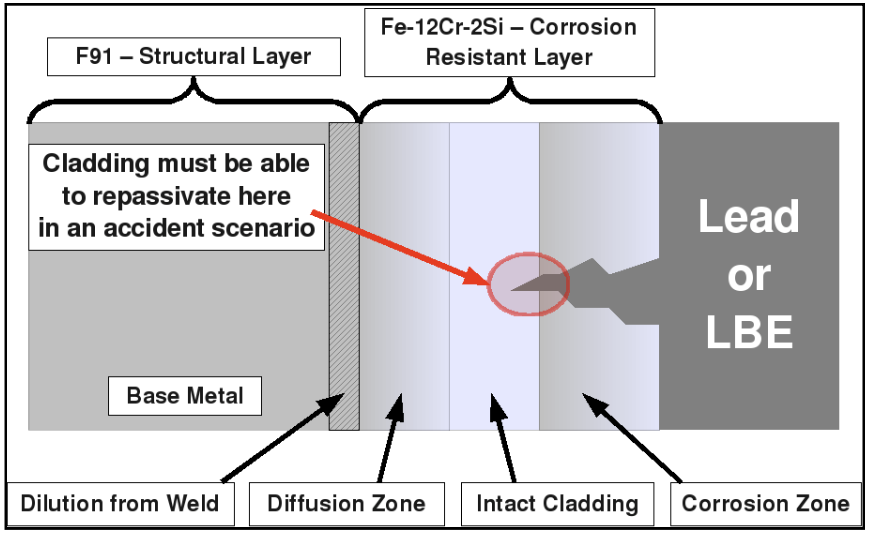
Concluding comments

- In spite of a rich history in LMFR development, current US efforts related to LFR are limited
- Nevertheless, there is continuing interest in LFR technology, mainly as a backup option to the SFR
- A small but dedicated group of researchers are continuing to maintain options through national lab, university and industry projects.

BACKUP SLIDES

MIT Work on Functionally Graded Composites (FGC)

- Objective: design and produce fuel cladding and coolant piping for HLM-cooled (Pb/LBE) fast reactors using commercial practices.
- The FGC consists of a structural layer of a 9Cr-1Mo steel (T/F91) and a corrosion resistant protective layer of a new alloy with a composition of Fe-12Cr-2Si.
- Extends Operating Temperature to 700° C and Flow Velocity to 6 m/sec
- The project proceeded to the point of production of Tube Reduced Extrusion (TREX) product for both OD and ID clad material.
- Follow-on will continue the development and further develop the properties data base.



Conclusions from MIT FGC investigations

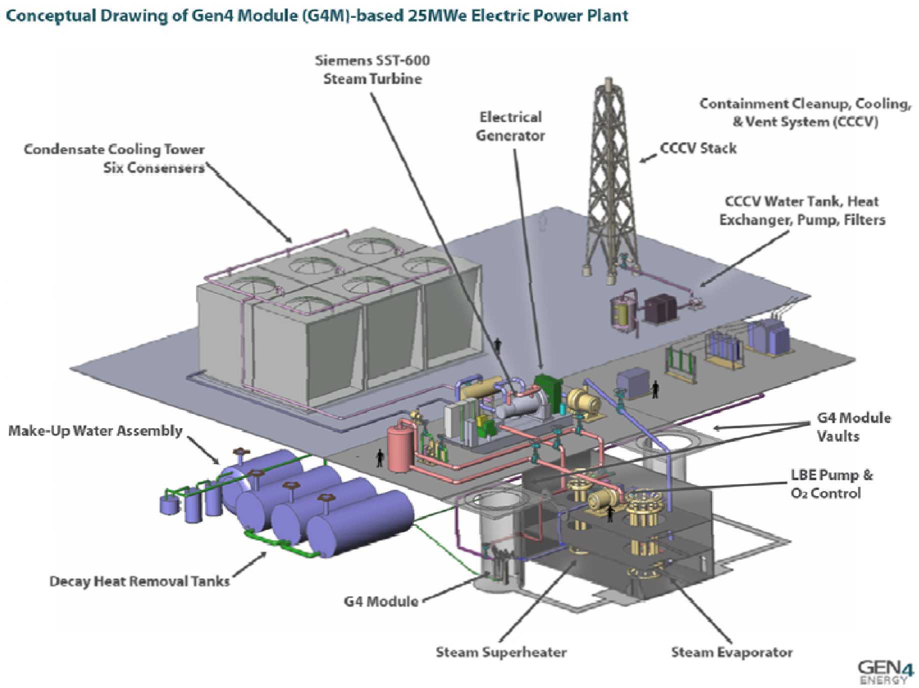
- The FGC protects against LBE corrosion in all expected environments, oxidizing or reducing, such that corrosion is no longer a concern for Pb/LBE-cooled systems.
- Extrapolated corrosion rates based on the experiments are less than 1 μm/yr , which is negligible for structural components, assuming a 60 year reactor lifetime.
- The FGC is diffusionally stable. The diffusional dilution zone between the two layers will not exceed 17 mm for fuel cladding (three year life) or 33 mm for coolant piping (sixty year life), both assumed to operate at 700C.
- Because of these performance gains, the FGC represents an enabling technology for Pb/LBE-cooled reactors and systems. A steady-state temperature increase of up to 150C beyond the current limitation of 550C is possible, provided that suitable structural materials exist.
- The FGC is ready for immediate deployment in non-irradiated or low-dose applications. The corrosion resistance has been demonstrated, and will be verified pending longer length experiments.

UC Berkeley continues with other materials and design work

- Use of the ICE2 experimental station to investigate the effects of irradiation on cladding steel corrosion in high temperature chemistry controlled heavy liquid metal environment (collaboration with LANL)
- Testing HT-9 steel in LBE at LANL's Ion Beam Materials Lab (IBML)
  - Introduced 2-4dpa on HT-9 at 430C and tested in LBE (~10-5wt% oxygen).
  - Tested sensor equipment under irradiation conditions.
  - PIE to be carried out at UC Berkeley during 2012.
- Testing commercial oxygen sensors in HLM
- Completed mechanical testing of oxide layers to determine the fracture stress needed to spall off the oxide layer
- Completed design feasibility studies of minimum burnup breed and burn (B&B) core concepts.

Gen-4 Energy is actively promoting its HPM system

Power	70MWt, 25MWe
Lifetime	8 – 10 years
Size (m)	1.5w x 2.5h
Weight (T)	Less than 50
Coolant	PbBi
Fuel	uranium nitride
Enrichment	<20%
Sealed Core	
Transportable with intact core	
Factory Fueled	

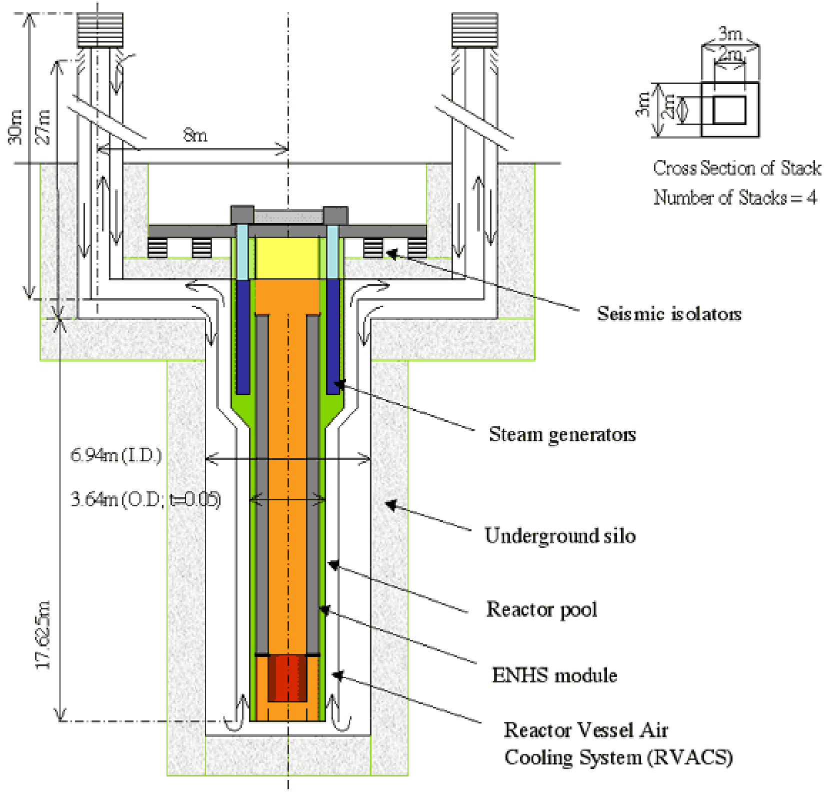


Sept 9, 2010: "Small modular reactor (SMR) start-up vendor Hyperion Power Generation has agreed to build a prototype mini-nuclear reactor at a U.S. Department of Energy laboratory. The company signed a memorandum of understanding with the Savannah River Nuclear Solutions to build the first demonstration reactor at the Savannah River Site (SRS) in South Carolina."

The STAR-ENHS (Encapsulated Nuclear Heat Source) concept was developed by a UC Berkeley-Led Team

- 3-year NERI study with UCB, ANL, Westinghouse, LLNL, KAIST and CRIEPI completed in FY02
- Evolutionary concept developed from CRIEPI-Toshiba 4S reactor
- Natural circulation cooling
- Reactor core heat transferred from primary to secondary Pb-Bi through capsule wall
- Fuel contained in capsule throughout fuel cycle
- Engineering feasibility demonstrated but economic feasibility is uncertain

Schematic vertical cut through the ENHS reactor



Lakechime PPRS is pursuing concepts described as “Evolutionary SSTAR”

- E-SSTAR
  - Evolutionary SSTAR variation intended to emphasize early deployment; currently at proposal stage. Features may include:
    - Forced cooling
    - Oxide fuel
    - Steam cycle power conversion
    - Small reactor