Progress of neutronics design on TMSR-SF1 and TMSR-LF1

Reactor Physics Division
TMSR SINAP CAS

Outline

- Introduction
- TMSR-SF1
- TMSR-LF1
- Some other issues
- Conclusion

TMSR roadmap

Long Term

- 100MW liquid-fueled demonstrated TMSR
- Th-U cycle demonstration

Medium Term

- 100MW solid-fueled demonstrated TMSR
- 10MW liquid-fueled experimental TMSR
- Commercial FHR demonstration

Near Term

- 10MW solid-fueled experimental TMSR(TMSR-SF1)
- 2MW liquid-fueled experimental TMSR(TMSR-LF1)
- Technologies Design Integration

Relationships between TMSR-SF and TMSR-LF

- Almost the same based technologies between TMSR-SF and TMSR-LF:
  - Salt loop technologies and Key equipments
  - Salt production and purification technologies
  - Materials and Measurment devices in high temperature salt condition
  - Energy utilization technologies
- Difference:
  - TMSR-SF: mechanical and physical reactor; TMSR-LF: chemical reactor
  - More radioactivity in the primary loop in TMSR-LF
  - TMSR-LF is more difficult than TMSR-SF based on current conditions

TMSR-SF1

- The first step is to build a test reactor (TMSR-SF1):
  - Demonstrate concept and safety of FHR.
  - Develop and integrate key technologies and components.
  - Build experimental platform for future reactors development: Reactor physics research, design, benchmark of modeling codes, technology and components test, materials and fuel (component) irradiated properties, etc..

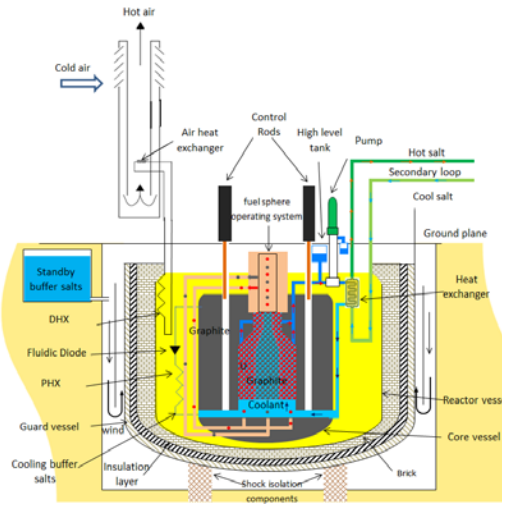
Basic design considerations

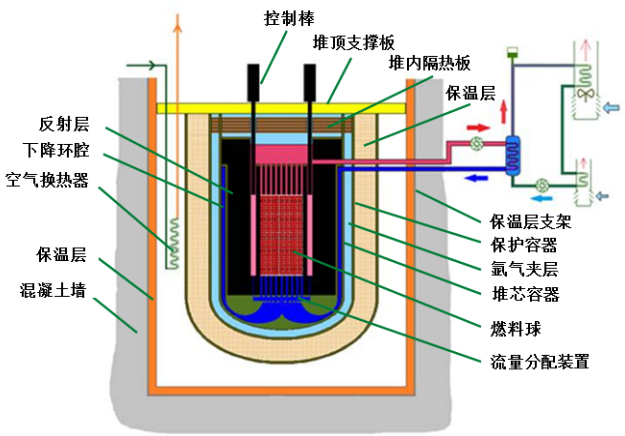
- **Safety** is most important issue. Designed by Using intrinsic safety characteristics, and with a large margin for safety parameters.
- **Feasibility**: Using existing materials, technologies as much as possible; Using simple design to achieve reliability.
- **Flexibility**: The reactor has designed with ability to cope with various uncertainties in material, engineering, data and model etc., and has relatively large tolerance.
- **Functions**: neutronics experiments， thermal-hydraulics experiments, safety experiments, materials and fuel test, component test, etc.

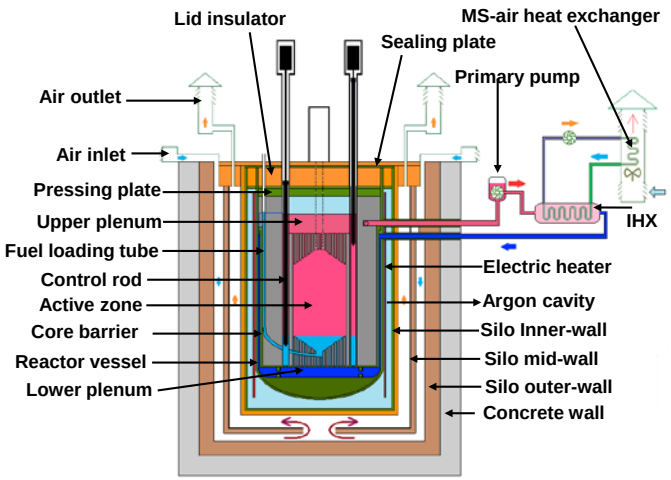
Main design features of SF1

- Reactor power: $10MW_{th}$
- Coolant temperature: Inlet 600°C, outlet 650°C.
- Fuel element: TRISO fuel, 6cm sphere.
- Core: Graphite core, conventional pebble bed arrangement.
- With passive residual heat removal.
- Temperature limitations: Fuel, <1400°C; coolant outlet, <750°C.
- Reactor vessel pressure limitations: <5atm.

The evolution of SF1 conceptual design







2012, 2MW

- Pre-conceptual design.
- Concept from literature.
- Too complicate and too much uncertainty.

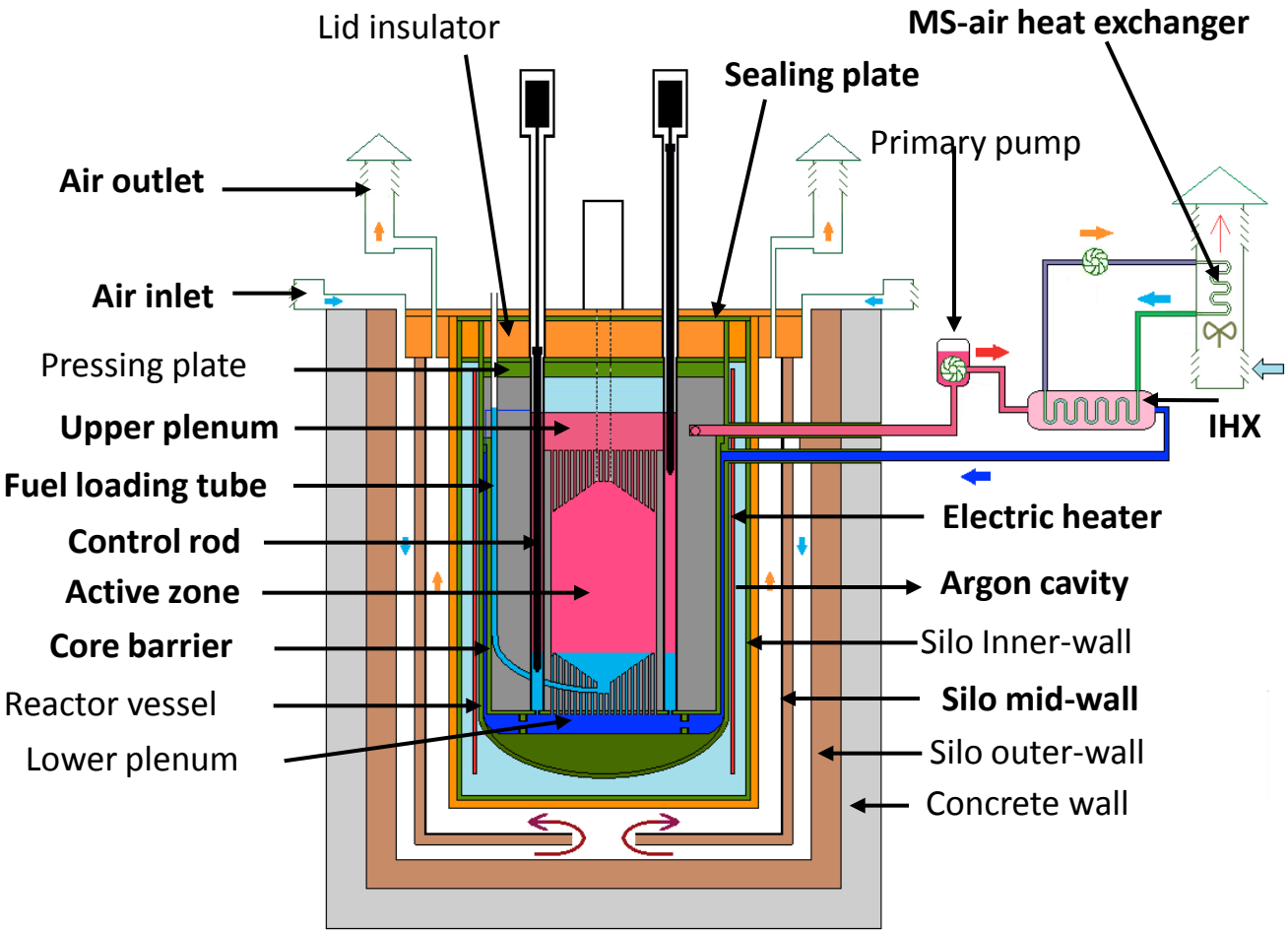
2013, 2MW

- Ordered pebble bed design.
- Accurate core model
- Little tolerance to uncertainties.
- Difficult for fuel loading and unloading.

2014, 10MW

- Conventional pebble bed design.
- Able to do reactor critical experiment.
- Future development on online fuel circulation.

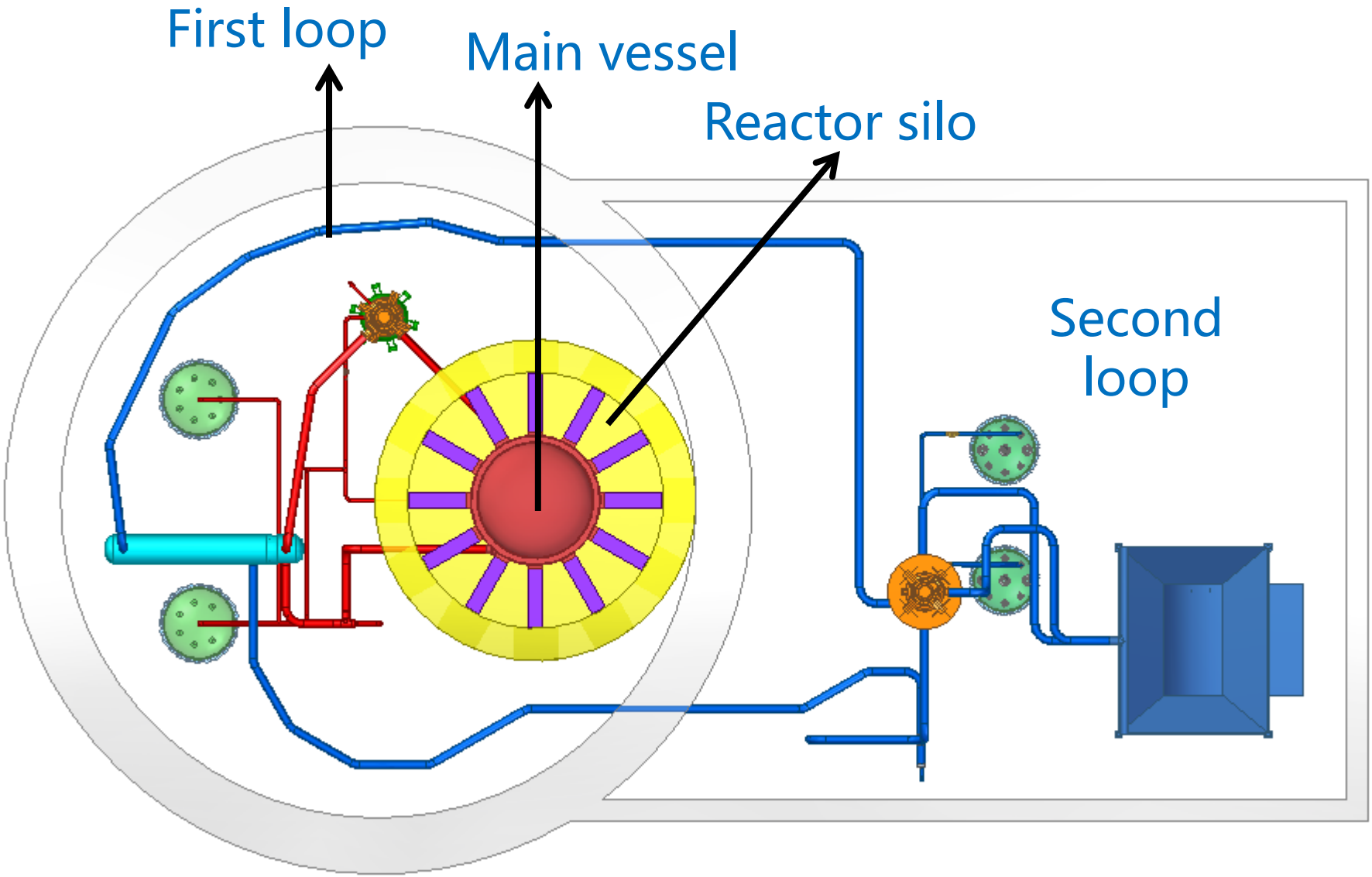
Schematic of SF1 main systems



Main systems

- Reactor core
- Coolant system: Primary loop and intermediate loop (Secondary loop).
- Main vessel, cavity, silo and shielding
- ...

Layout of reactor



Choice for decay heat removal pool

- In the first version of TMSR-SF1, there is a decay heat reomoval pool and a DRACS-PRACS system.

- In the next versions of TMSR-SF1, decay heat removal system changed to a passive "core-vessel-silo-air cooler" system as the following reasons: (1) more simpler; (2) more reliable; (3) 10MW power and low power density means its feasibility.

Molten Salt

- Primary loop: FLiBe molten salt.
  - ^{7}Li enrichment~99.99%. Enrichment of 99.914% have been demonstrated in lab.
  - Contaminant boron equivalent: 2ppm.
- Secondary loop: FliNak molten salt.
- Cover gas: Argon.

Choice for concentration of Li-7

- 99.995% concentration of Li-7 is a good choice for initial reactor design. Total neutron absorption proportion of Li-7 and Li-6 are about equvalent at this condition.

- Because of time schedule limitation for li-7 purification, 99.99% concentration is choosed for TMSR-SF1 design, which is worse on critical and temperature reactivity coefficient.

Structure materials

- Alloy-N: Reactor vessel, in core structure, Primary loop.
  - Exp. data up to 800°C have been collected for structure design and structure safety assessment of reactor and loop.
- Graphite: Reflector
  - Alternative: fine grain graphite under development, resistant to molten salt up to 0.6MPa pressure.
  - Neutron irradiation test underway.
- C-C composite: In core guide tubes for control rods.

Fuel element

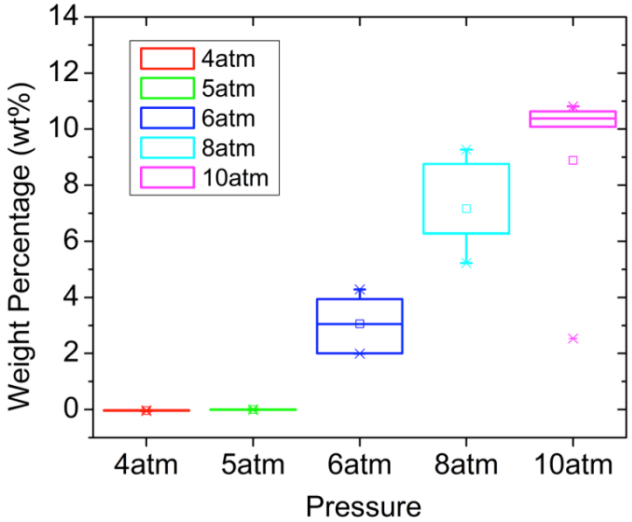
- SF1 will use fuel elements developed by INET Tsinghua University for HTR-PM, except U5 enrichment increased to 17% (which is the same as HTR-10);
- Fuel geometry: 6 cm diameter sphere;
- Uranium loading : 7g;
- Density of fuel element: 1.81g/cm^3. (about 93% of molten salt density)

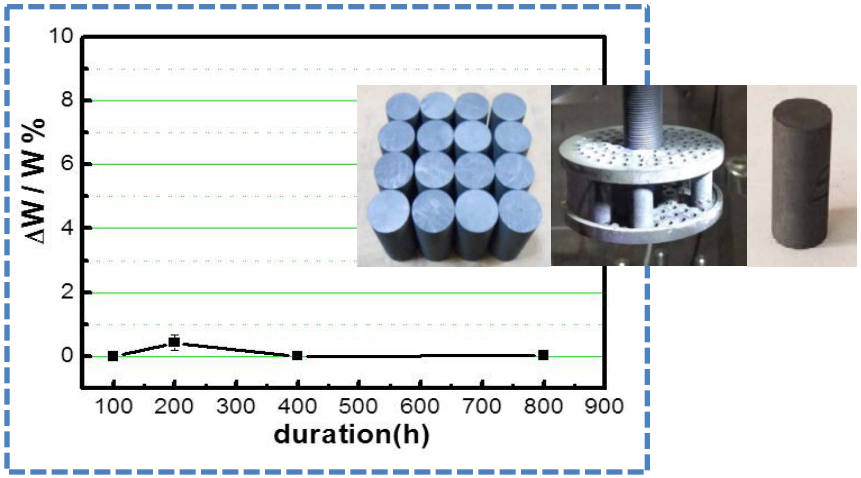
Fuel qualification

- The designed fuel failure temperature is higher than 1600°C.
- The irradiation test of HTR-PM fuel element completed recently at HFR, Holland.
  - Average temperature at fuel element center: 1050°C;
  - Maximum burn up: 113MWd/kgHM.
  - Fission gas release rate: ~10^{-9}.
- Much better than SF1 designed parameters.

Fuel qualification

- Risk of molten salt penetration into fuel elements?
  - Increase density and salt attack fuel particle.
  - **Experiments show unirradiated graphite shell is molten salt resistant.**





The behaviour of irradiated shell material may be different.

Same experiments for irradiated shell material are required.

Fuel management

📖 **Three designs for Fuel management:**

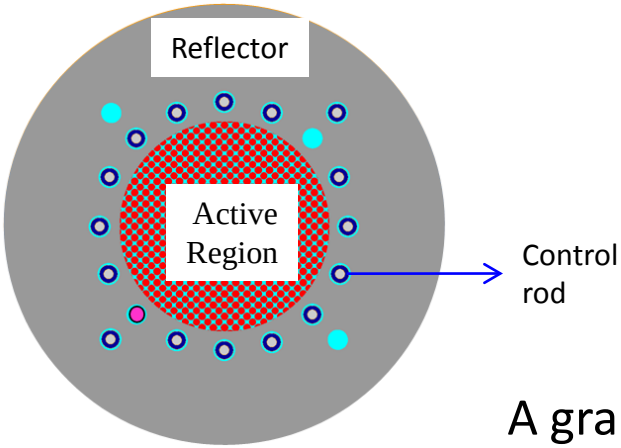
1. **On-line loading and on-line defuel**
2. **Three batcher loading and one batch defuel**
3. **One batch loading and one batch defuel**

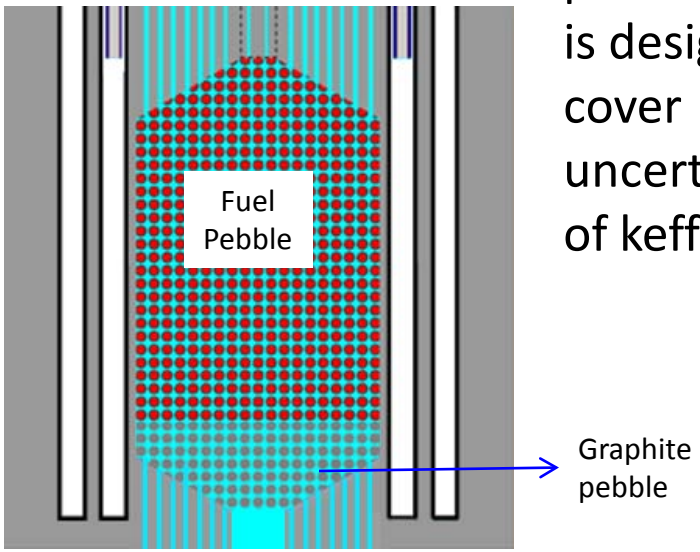
- The third one is the current basis design for considerations as the following reasons: (1) more simpler; (2) more safety; (3) enough for experiments.

- The first one and the second one are two alternative vesions

Schematic core design

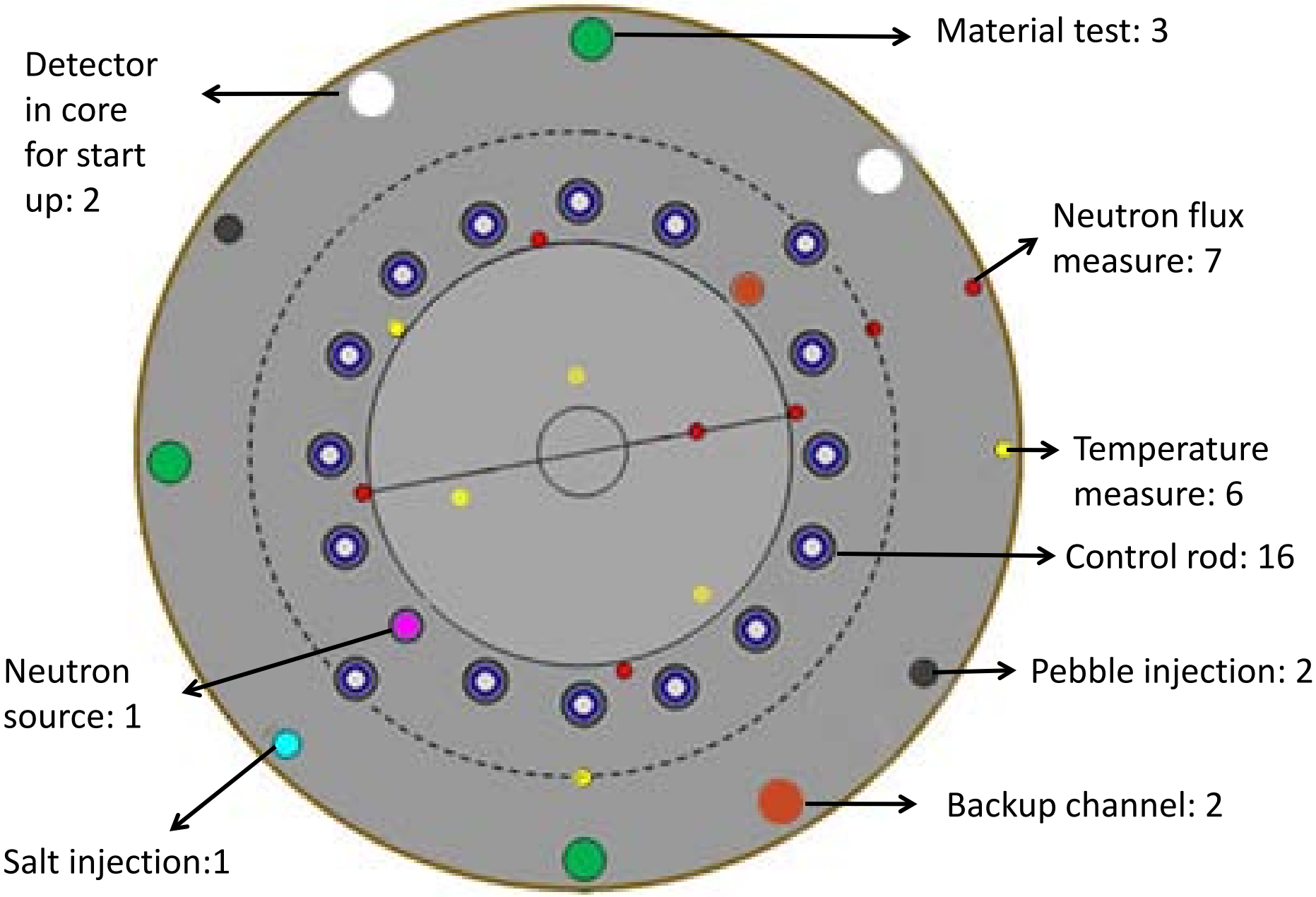
Parameter	Value
Structure of active area	truncated cone+Cylinder
Height of Cylinder active area	180 cm
Cylinder diameter of active area	135cm
Angle of upper and lower cone with horizontal	30°
Height of reflector	300 cm
Outer diameter of reflector	285cm
The thickness of radial reflector	75 cm
Pebble bed packing fraction	60%





A graphite pebble region is designed to cover uncertainties of keff.

Core layout


Material test: 3
Detector in core for start up: 2
Neutron flux measure: 7
Temperature measure: 6
Control rod: 16
Neutron source: 1
Pebble injection: 2
Backup channel: 2
Salt injection:1

Shutdown systems

- **Two Shutdown systems**
- **The first one**: 13 control rods (8 shim rods, 2 regulating rods, 3 safety rods)
- **The second one**: 3 safety rods (different driving mechanism) + drain salts in core

EFPD	Worth of the first Shutdown system (pcm)	
	Full power (625℃)	Cold state (550℃)
0.0	11378	10719
2.0	11592	10934
50.0	11579	11124
100.0	11729	11236

EFPD	keff after inserting the 3 safety rods of the second shutdown system
0.0	0.98438±0.00042
2.0	0.97900 ±0.00040
50.0	0.97628 ±0.00040
100.0	0.97469 ±0.00040

EFPD	keff after draining salts
0.0	0.84305 ±0.00044
2.0	0.88201 ±0.00045
50.0	0.89643 ±0.00042
100.0	0.90725 ±0.00045

Main parameters

EFPD	100 days
Diameter and number of sphere	6.0cm / 13500
Enrichment and massof ^{235}U	17.0% / 15.6kg
Cold state / reactivity	550℃ / 7200pcm
The first shutdown system and its margin	13 control rods/ ~2600pcm
The second shutdown system and its margin	3 control rods+drain salts / ~10000pcm
Temperature reactivity coefficient (BOL、EOL)	-5.54pcm/K -3.66pcm/K
Max burnup	16000 MWd/ tU
Mass of Tirtium (one EFPD)	2.4 g

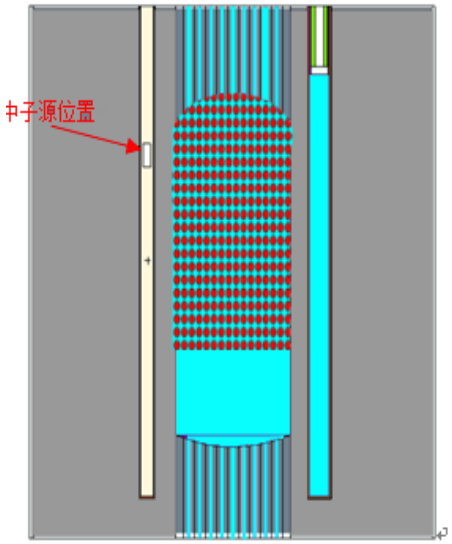
Main parameters - 2

Power density	3.9 MW/m3
Power peak factor（BOL、EOL）	1.53 / 1.50
Total TRC（BOL、EOL）	-5.54pcm/K, -3.66PCM/K
Fuel TRC（BOL、EOL）	-2.40pcm/K, -2.59PCM/K
Coolant TRC（BOL、EOL）	-0.47pcm/K, +0.08PCM/K
Graphite in Pebble TRC（BOL、EOL）	-3.95pcm/K, -2.64PCM/K
Reflector TRC（BOL、EOL）	+1.12pcm/K, +1.51PCM/K

Neutron source

- A Am-Be neutron source is used in physical start-up to avoid dead zone of monitoring, especially in the first physical start-up .
- Neutron source is in channel in graphite reflector
  - Am-Be neutron source intensity: 20Ci, $4.4*10^7$n/s

Source type	Energy of neutron MeV	Half-life period	yield 10^6n/(s*10^{10}Bq α)
^{241}Am-Be	4.2-5	432 years	0.60-0.74
^{210}Po-Be	4.2	138 days	0.67-0.80
238Pu-Be		88 years	
Cf252	2.3	2.65 years	$2.31*10^{12}$ n/s/g



Detectors of neutronics

- **Four kinds of detectors is used to monitor neutron flux (power) in real time.**
  - 3 Source range detectors in reflector
  - 2 Source range detectors out of main vessel
  - 2 medium range detectors out of main vessel
  - 3 power range detectors out of main vessel
- **Two kinds of methods is used to measure neutron flux distribution and spectra.**
  - Used multi metal foils to measure neutron flux and spectra in different small channel in core.
  - Used small detectors to measure neutron flux in different small channel in core.

Source term –Tritium

Source and consideration :

- Li is the main source of tritium in the primary loop. The corresponding reaction equation is :

$n + {}^{6}Li \Longrightarrow {}^{3}H + {}^{4}He$

$n + {}^{7}Li \Longrightarrow {}^{3}H + {}^{4}He + n$

$$\frac{dN_{Li6}(t)}{dt} = N_{Be9} \bullet \phi \bullet \sigma(n,\alpha) - N_{Li6}(t) \bullet \phi \bullet \sigma(n,T)$$

$$\frac{dN_T(t)}{dt} = N_{Li6}(t) \bullet \phi \bullet \sigma(n,T) - \lambda N_T(t)$$

$$N_T(t) = \phi\sigma(n,T)\left(N_{1i6}(0) - \frac{N_{Be9}\sigma(n,\alpha)}{\sigma(n,T)}\right) \bullet \frac{e^{-\phi\sigma(n,T)t} - e^{-\lambda t}}{\lambda - \phi\sigma(n,T)} + \frac{N_{Be9}\sigma(n,\alpha)}{\sigma(n,T)} \bullet \phi\sigma(n,T) \bullet \frac{1 - e^{-\lambda t}}{\lambda}$$

Note：N：atomic density σ：micro-cross section φ：Neutron flux λ：decay constant

Total mass or tritium is about 2.4 g at EOL

Source term - C14

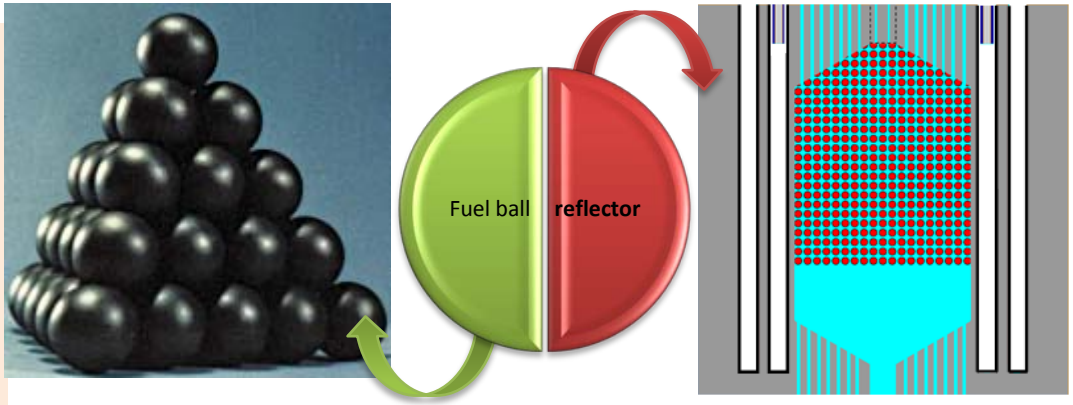
Purposes： C-14 is one of main radioactive source of TMSR-SF， half-time is 5730a, there are clear emissions limits of C-14 for different type of reactor in" Nuclear power plant environment radiation protection regulations". Have to consider the production of C14, Provide input for the analysis of other systems.

Source and calculation:

N – 14 impurity in primary circuit is the main source of C -14 compared with O-17 impurity and fuel pebble shell wear.

Related reaction：

$^{14}N + n \rightarrow {}^{1}H + {}^{14}C$ cross section is **0.29b**



analysis：

N - 14 impurity in primary circuit is the main source to the C -14 contribution， which contribution is 4 orders higher than other way such as O-17 impurity and fuel pebble shell wear($^{13}C(n,\gamma)^{14}C$).

C-14 total activity in primary circuit is 1.24E+10Bq.

TRC (Temperature Reactivity Coefficient)

- **TRC is a very important safety parameter， include Total TRC,Coolant TRC, Fuel TRC, Moderate TRC**

- **TRC is different in different conditions (especially Coolant TRC)**
  - Infinite lattice cell and finite core
  - With control rods and without control rods in core
  - According to different leakage proportion of neutron from core
  - According to different burnup
  - According to different enrichment of Li-6
  - Etc.

TRC analysis with four factor formula

$$k_\infty = \varepsilon p \eta f$$

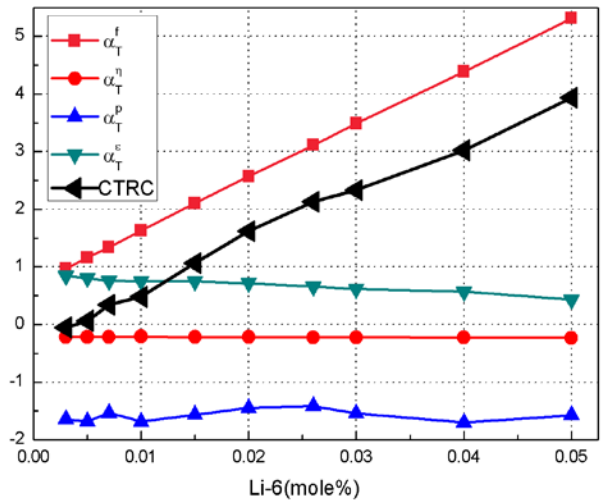
📖 Case 1: Coolant TRC with different enrichment of Li-6 in infinite lattice

^{6}Li（mole%）	α (f)	α (η)	α (p)	α (ε)	(pcm/K)
0.003	9.671E-06	-2.136E-06	-1.646E-05	8.416E-06	-0.0510
0.005	1.158E-05	-2.139E-06	-1.681E-05	8.013E-06	0.0651
0.007	1.338E-05	-2.134E-06	-1.537E-05	7.539E-06	0.3418
0.01	1.628E-05	-2.084E-06	-1.687E-05	7.473E-06	0.4804
0.02	2.571E-05	-2.181E-06	-1.449E-05	7.145E-06	1.6184
0.03	3.487E-05	-2.231E-06	-1.543E-05	6.107E-06	2.3312
0.04	4.385E-05	-2.272E-06	-1.704E-05	5.690E-06	3.0232
0.05	5.314E-05	-2.309E-06	-1.577E-05	4.293E-06	3.9349
	Large positive	**No change** **Small negative**	**No change** **negative**	**Small positive**	

•SUN Jianyou, **ZOU Yang** and XU Hongjie, etc. Analysis of the coolant reactivity coefficients of FHRs with 6Li contents of coolant, Nuclear Science and Techniques, Vol 37(9), pp 090605(1-7), Sep, 2014. (Chinese version)

Effect of Li-6 Concentration

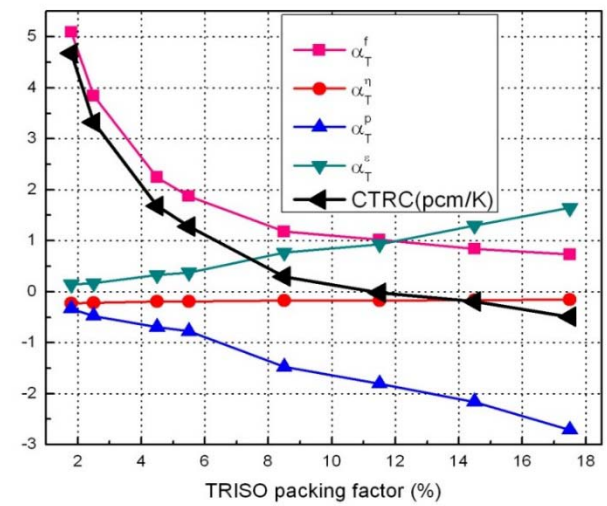
- High Li-6 concentration makes f dominate.



Case 2: different Triso packing factor.

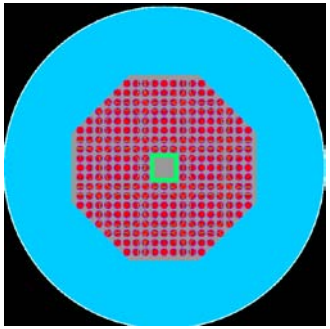
Effect of Neutron Spectrum

- f is mainly positive contributor;
- p is mainly negative contributor;
- Soft spectrum makes f dominate.



Case 3: Coolant TRC with different neutron leakage (equvalent to different core size)

$$k_{\text{eff}} = k_{\infty}\Lambda = \varepsilon p \eta f \Lambda$$

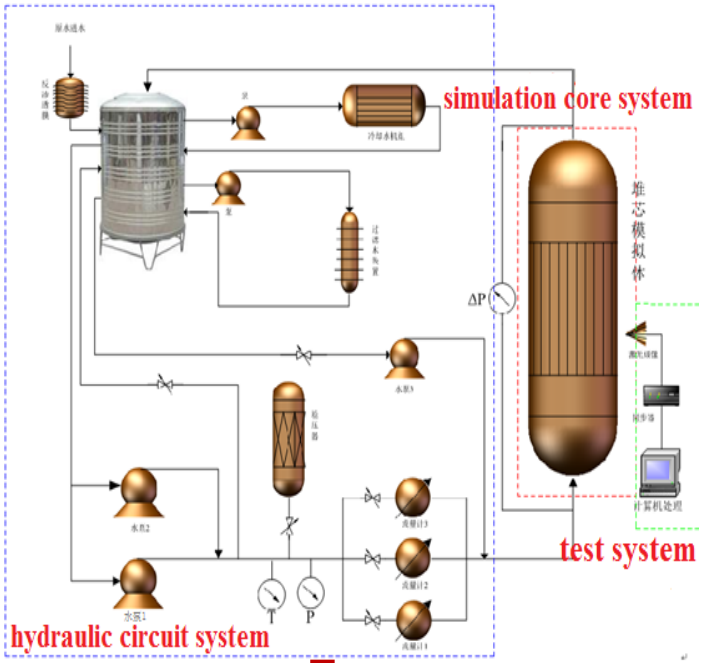


^{6}Li (mole%)	α (f)	α (η)	α (p)	α (ε)	α (Λ)	(pcm/K)
2.14	-2.26E-06	-2.22E-06	-7.35E-06	7.23E-06	-8.79E-06	-1.3397
3.776	9.08E-07	-2.21E-06	-7.87E-06	6.76E-06	-7.12E-06	-0.9535
8.886	4.70E-06	-2.05E-06	-9.68E-06	6.79E-06	-4.66E-06	-0.4884
34.145	8.01E-06	-1.97E-06	-1.17E-05	7.08E-06	-2.90E-06	-0.1443
308.305	9.16E-06	-1.79E-06	-1.30E-05	7.56E-06	-1.11E-06	0.0808
963.905	1.12E-05	-1.75E-06	-1.49E-05	7.37E-06	-4.27E-07	0.1481
∞	1.14E-05	-1.75E-06	-1.33E-05	7.45E-06	0	0.3805
	Large positive	**No change Small negative**	**Change to more negative**	**Small positive**	**Small negative in finite core**	

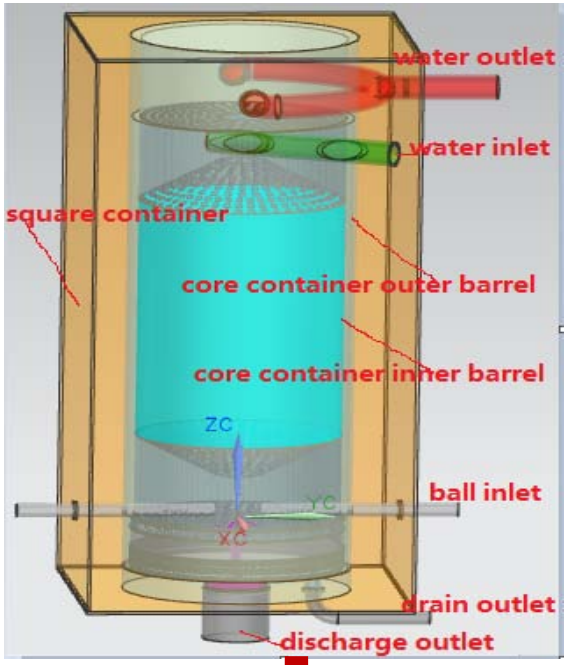
•SUN Jianyou, **ZOU Yang**, XU Hongjie, etc. Study on the influence of core volume of PB-FHR on coolant temperature reactivity coefficient, Nuclear Science and Techniques, Vol 37(12), pp 120603(1-7), Oct, 2014. (Chinese version)

PIV pebble-bed experimental facility

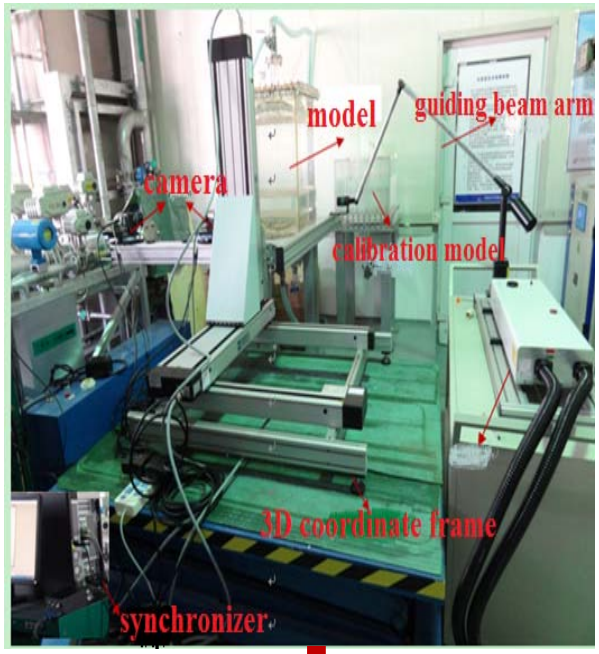
- Distribution of pebble spheres in core is important for physics and thermal-hydraulics analysis.
- SINAP is building a facility to record position of every sphere
- Medium: NaI solution; Sphere: PMMA polymer ball



Loop



simulation core



measure system

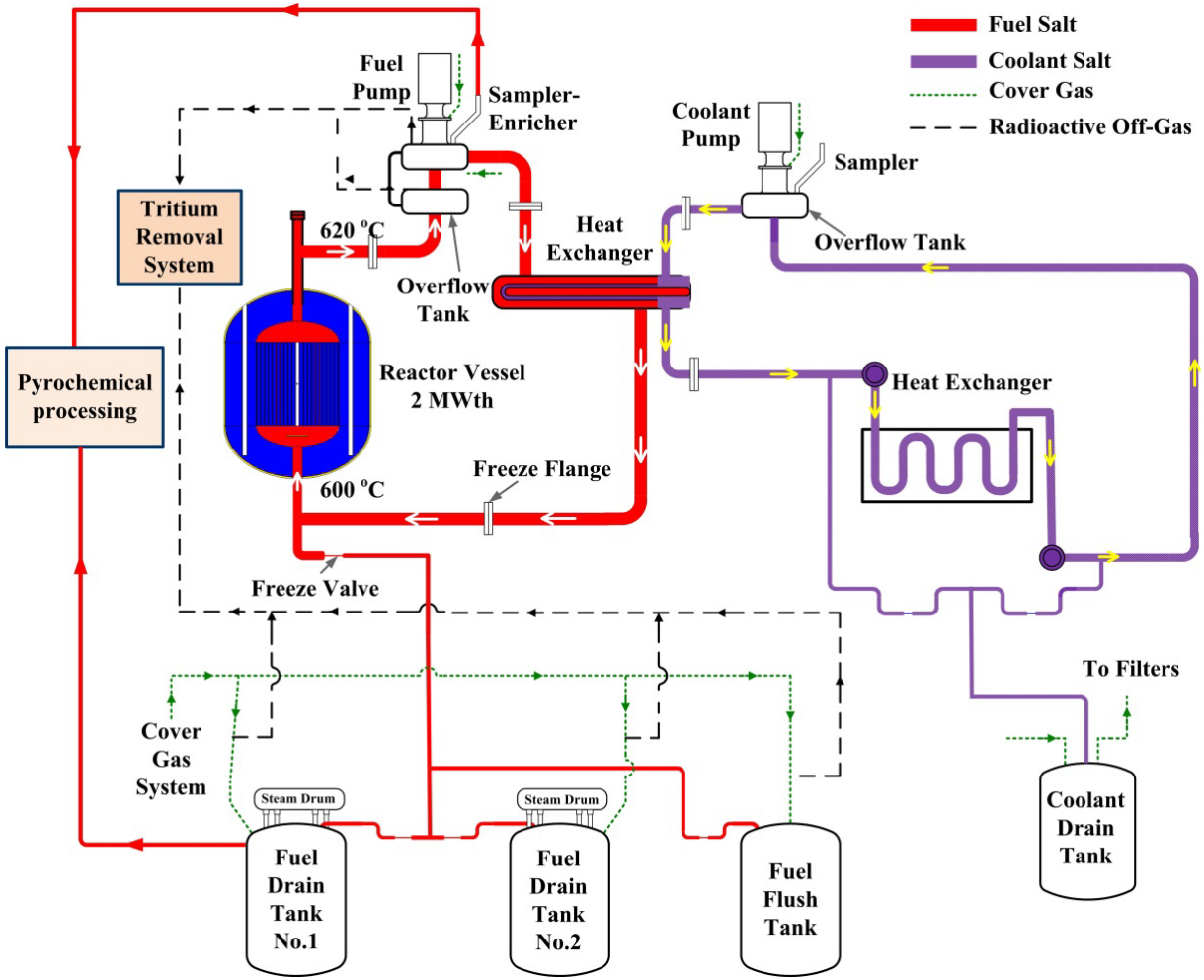
TMSR-LF1

- Based on technologies of TMSR-SF1 and pyroprocessing, TMSR-LF1 (2 MW) will be built.
- Demonstrate concept of MSR with liquid fuel and pyroprocessing.
- Demonstrate Th-U cycle and its features.
- MSR physics and chemistry research.
- Build experimental platform for future reactors development.

Basic design considerations

- U-235 enrichment less than 20%; (different constrains with MSRE)
- Li-7 abundance: 99.95% ; (lower requirement than SF)
- Appropriate amount of Th (demonstration for Th-U fuel cycle and pyroprocessing) ;
- Low excess reactivity; (virtue of MSR)
- Negative temperature feedback; (for safety)
- Simple core structure, easy for system integration, control and maintenance;
- cope with various uncertainties in physics design

TMSR-LF1 layout



- Reactor core
- Heat transfer system
- Pyroprocessing
- Radioactive gas removal system
- Safety system
- Auxiliary components

Main design features of LF1

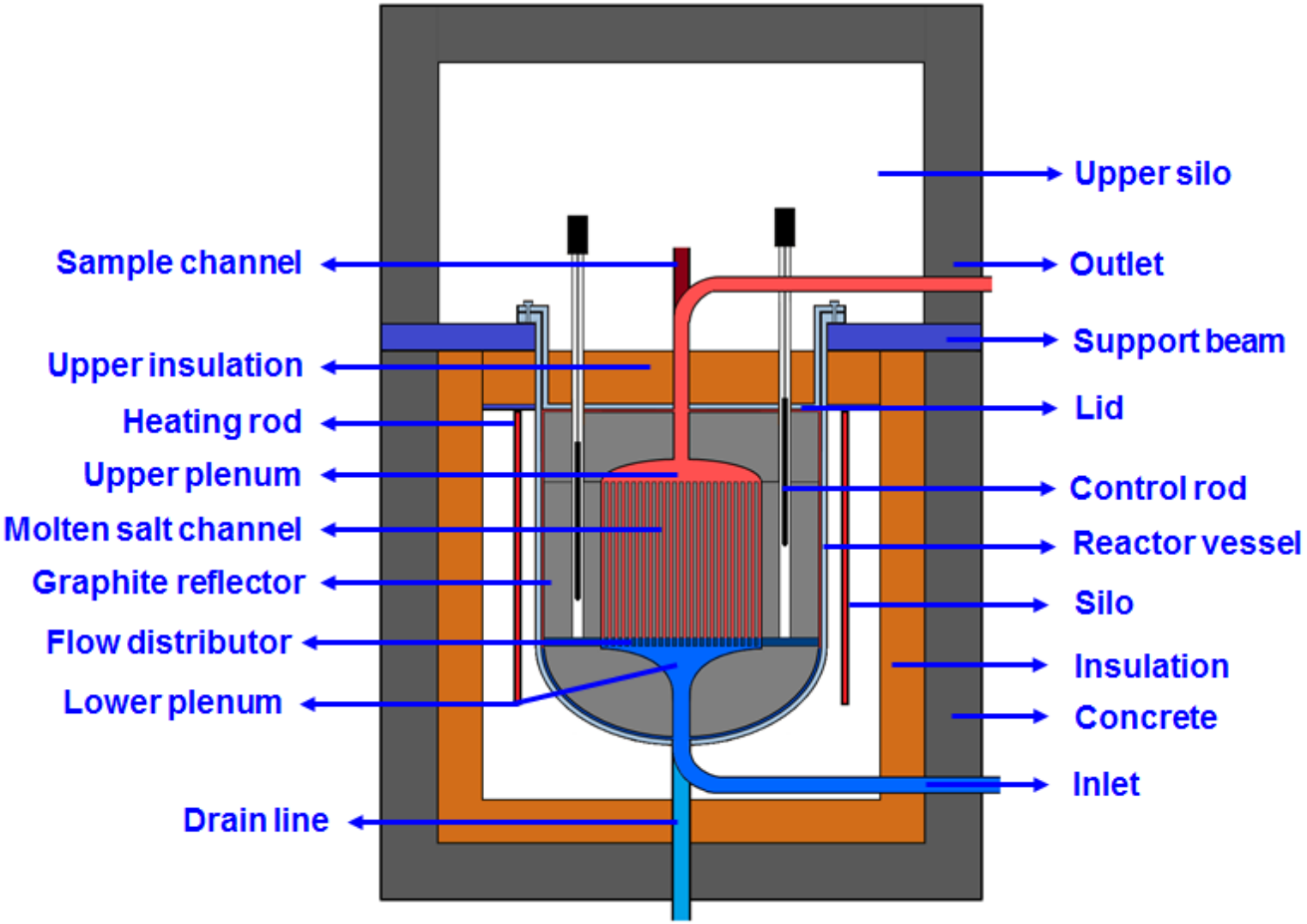
- Reactor power: $2MW_{th}$
- Coolant temperature: Inlet 600°C, outlet 620°C.
- Fuel element: LiF-BeF2-ThF4-UF4 (mol%)
- Core: Graphite core
- With passive residual heat removal.
- Reactor vessel pressure limitations: <5atm.

Fuel salt candidates for TMSR-LF1

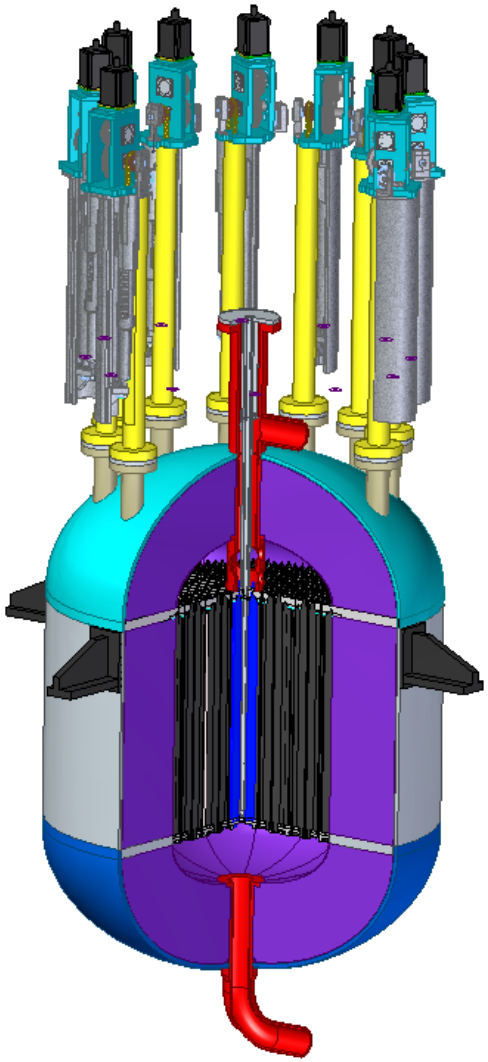
- Fuel solubility must meet the critical requirement
- Good physical properties such as vapor pressure, viscosity
- Proper charge potential (U4+/U3+) during running
- O2- concentration should be limited to decrease UO2/ThO2 precipitation

Fuel and coolant salts	
$LiF-BeF_2-ThF_4-UF_4$ (mol%)	**Melting point(℃)**
68-29-0.22-2.78	470
LiF-NaF-KF	**Melting point(℃)**
46.5%-11.4%-42%	454

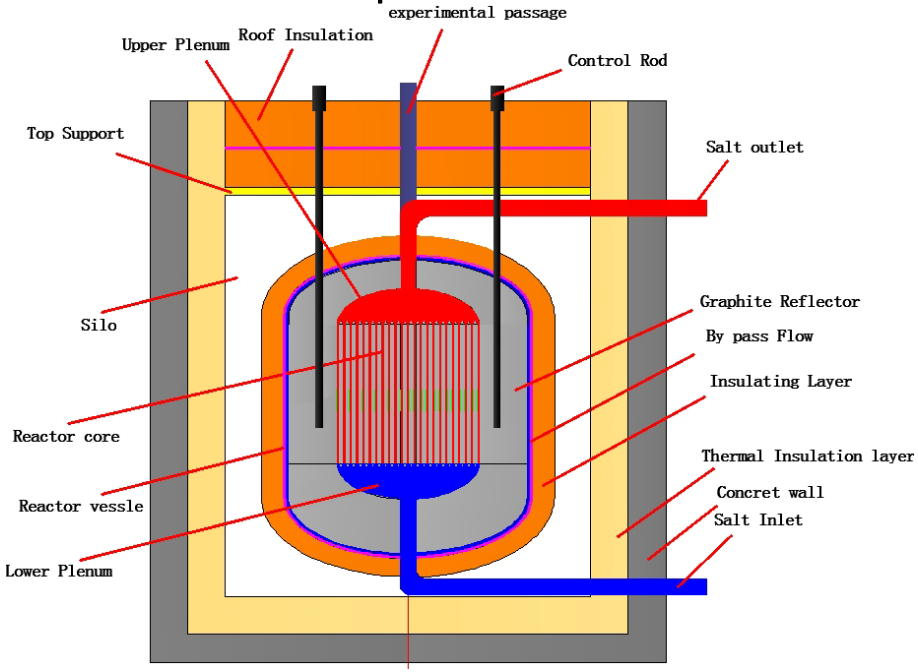
Reactor vessel and silo



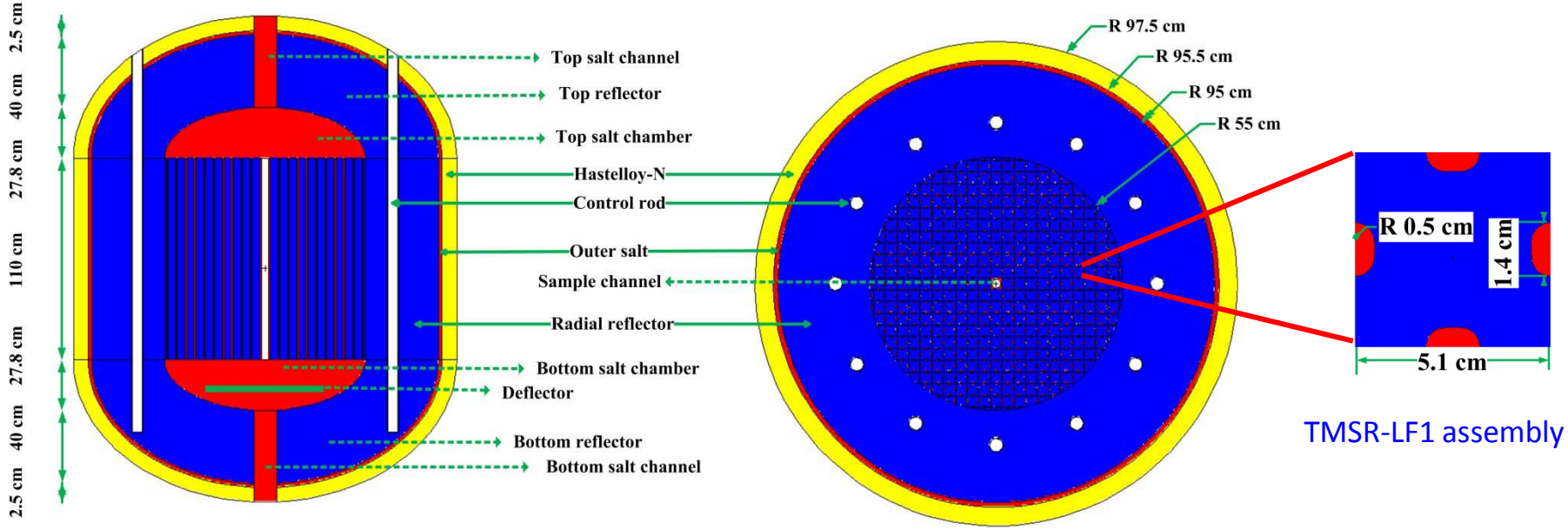
Reactor layout



- Reactor core
- Reactor vessel
- Graphite moderator and reflector
- Control rods
- Primary salt loop
- Sample channel



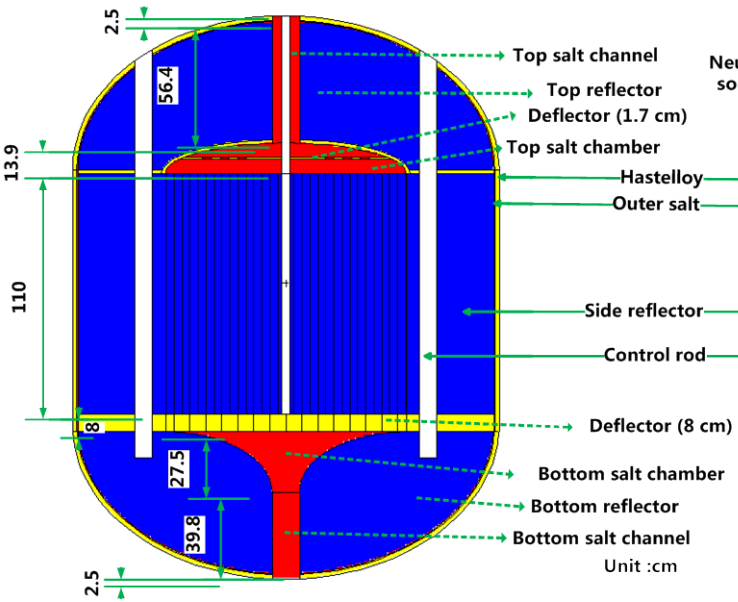
Core design parameters for TMSR-LF1

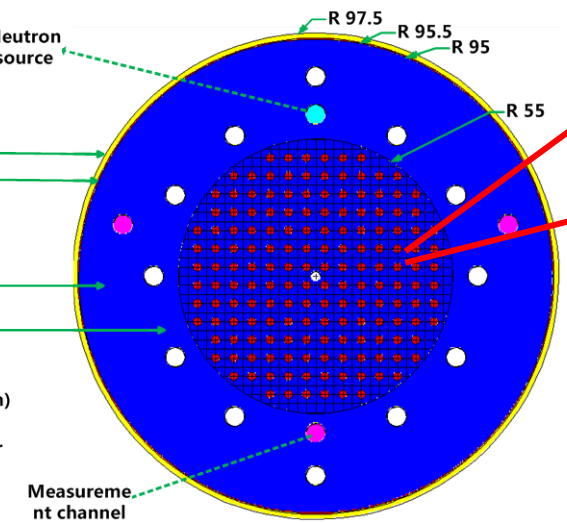


TMSR-LF1 assembly

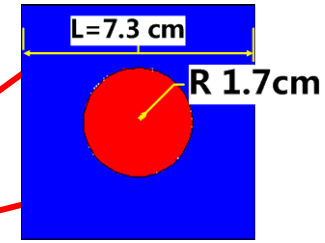
Parameters	Value	Parameters	Value
Thermal power	2 MWth	Core diameter / height	195/251 cm
Active core diameter / height	110/110 cm	Reflector thickness	40 cm
Total fuel salt volume	0.54 m^3	Total reactivity coefficient	-9.0 pcm/K
Fuel salt inlet temperature	600 ℃	Fuel salt outlet temperature	620 ℃
control rods	12		

Modified core design parameters for TMSR-LF1

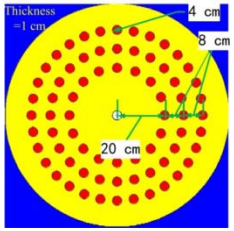




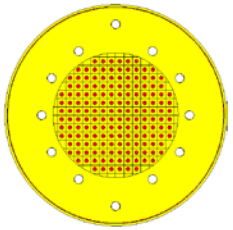
TMSR-LF1 assembly



Top deflector



Bottom deflector



Parameters	Value	Parameters	Value
Thermal power	2 MWth	Core diameter / height	195/251 cm
Active core diameter / height	110/110 cm	Reflector thickness	~40-55 cm
Total fuel salt volume	0.385 m^3	Total reactivity coefficient	-6.0 pcm/K
Fuel salt inlet temperature	600 ℃	Fuel salt outlet temperature	620 ℃
control rods	12		

Comparison of design parameters of two versions

Parameters	**Initial version**	**Current version**
Reactor diameter / height, cm	195/251	195/251
Core diameter / height, cm	110/110	110/110
Axial reflector thickness, cm	40	40-55
Fuel salt volume in the active zone, m^3	0.096	0.184
Total fuel salt volume in core, m^3	0.538	0.385
Fuel salt Composition mol%	$LiF-BeF_2-ThF_4-UF_4$	$LiF-BeF_2-ThF_4-UF_4$
	70.2-26.8-0.073-2.927	68-28-0.1-3.9
Li-7 abundance, %	99.95	99.95
U-235 enrichment, %	16.2	18.6
U-235 loading, kg	33	42.04
Th-232 loading, kg	5.0	5.66
Average operation temperature, ℃	610	610
Temperature reactivity coefficient, pcm/K	-9.0	-6.0
Control rod worth (@610 ℃) , pcm	5832	9038
CR(Th-232) @BOL	0.0093	0.0099
CR(U-238)@BOL	0.2080	0.1991

Th-U conversion optimized in the future

Considerations on increaseing thorium inventory:

- ☐ To meet the requirements of Th-U conversion demonstration
- ☐ U-235 enrichment less than 20% (considering online refueling)
- ☐ Low heavy nuclei inventory (low requirement for oxygen control)

Increase HN

- **Negligible effect on Th-U conversion**
- **Weaken the temperature feedback**

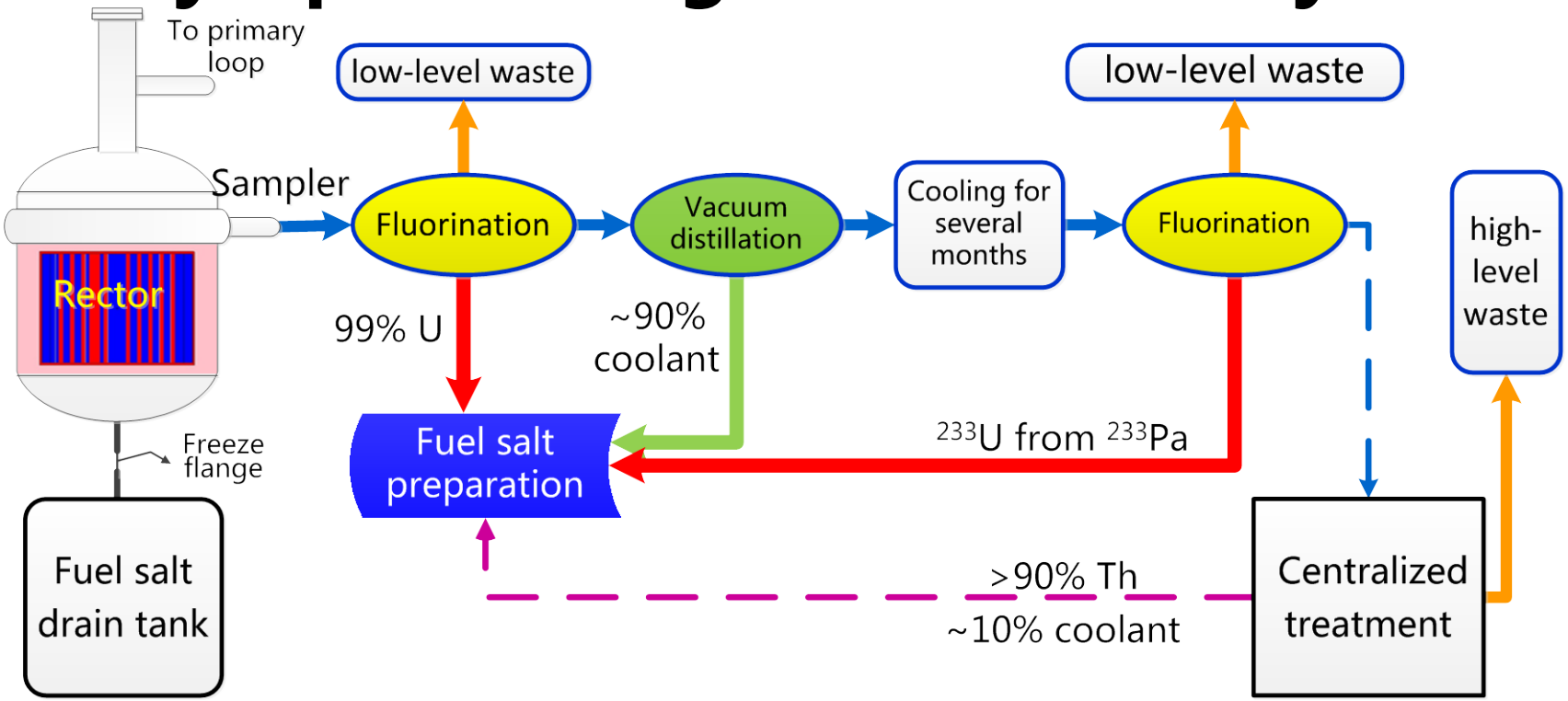
increase Li-7 enrichment

- **Improve Th-U conversion**
- **Decrease tritium production**

Increase U-235 enrichment

- **Negligible effect on Th-U conversion**
- **Limited Th inventory (due to U-235 <20%)**

Pyroprocessing of Th-U fuel cycle



- Sampler~ 10L/d;
- Fuel salt batch reprocessing: 10 L per course
- Thorium ~ 50 kg in core;

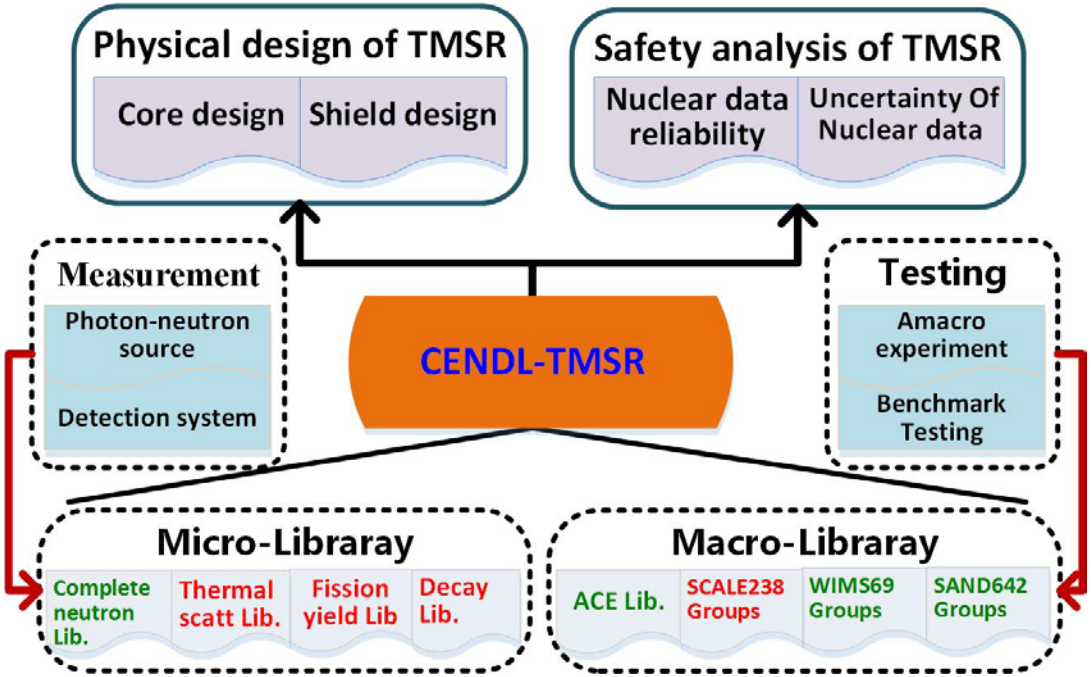
- demonstration of key technologies: reprocessing flowsheet, fluorination and Vacuum distillation
- Validation of Th-U conversion and online refueling

Some other issues

- Nuclear database
- Thermal neutron scattering cross-section of FLiBe
- Delay neutron from fission production and photoneutron

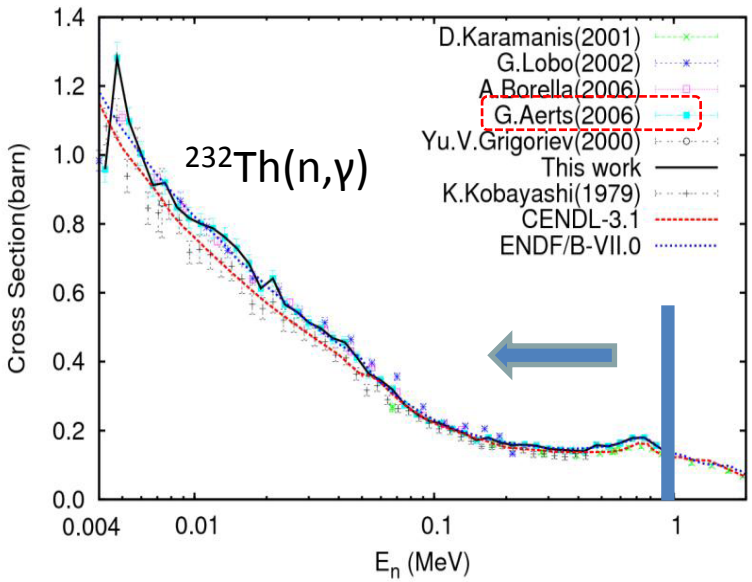
Nuclear database

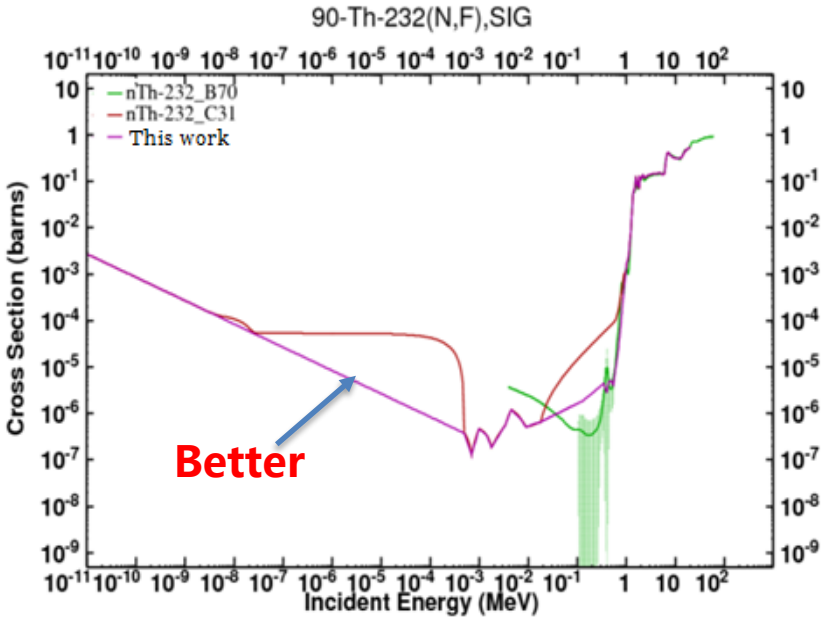
- According to requirement of nuclear data in TMSR project , TMSR is building a Th/U cycle special nuclear data library (CENDL-TMSR) which is a sub-library in CENDL Lib.
- Complete the evaluate system of Th/U cycle nuclear data ,and obtain the more accurate nuclear data for physical design of TMSR until 2030 .



CENDL-TMSR-V0

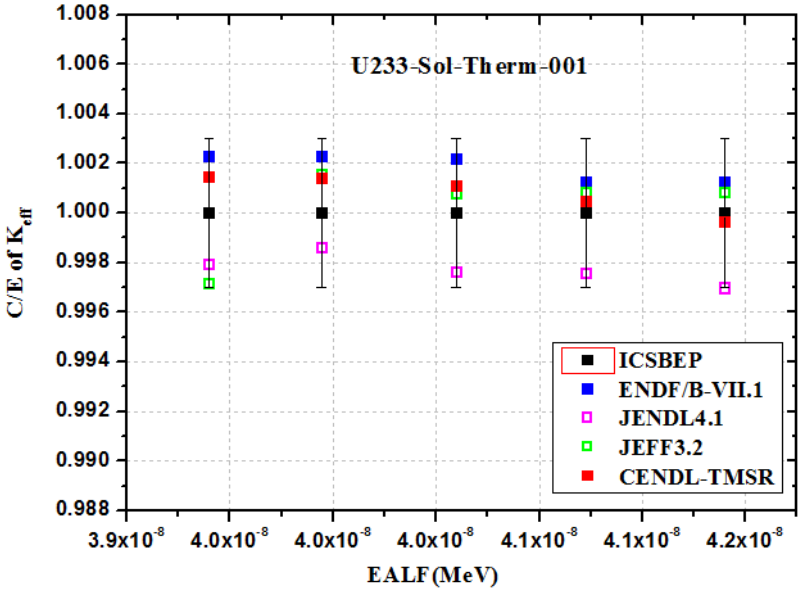
- Re-evaluate nuclear data for ^{232}Th、^{233}U、^{7,6}Li and other key nuclides.
- Add the thermal neutron scattering data of molten salt.
- Build the CENDL-TMSR-V0 evaluated nuclear data which is based on re-evaluated nuclear data and the result of the benchmark testing for ENDF/B、JENDL、JEFF and CENDL.

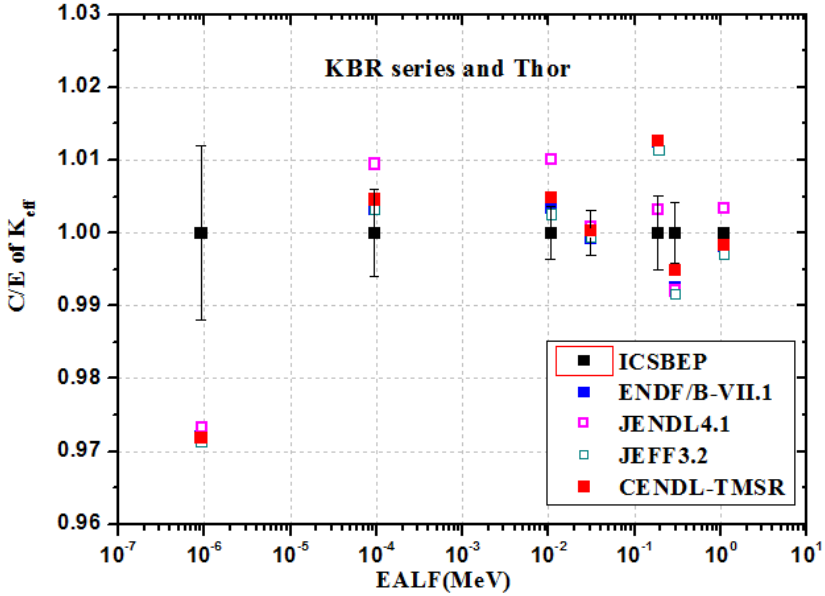




Benchmark tests for CENDL-TMSR-V0

- **Data source** ： ENDF/B-VII.1, JENDL4.1, JEFF3.2, CENDL-TMSR.
- **Temperature：** 300K.
- **Critical benchmark experiments:** Based on ICSBEP2006, include U233-SOL-THERM for ^{233}U, the KBR series and Thor for ^{232}Th and so on.
- **Conclusion :**From the results of benchmark testing, CENDL-TMSR-V0 has reached the level of ENDF/B-VII.1, and part of the benchmark devices are slightly better.





Thermal neutron scattering of FLiBe

- Thermal neutron scattering cross section have a large influence on critical analysis of MSR. Evaluateing this effect of liquid FLiBe is important.
- According to raw estimation, the influence of thermal neutron scattering of liquid FLiBe molten salt on K_{eff} in MSRE is about 0.5~1% (700~1200K).

Material behavior and structure (Crystal, Liquid, Gas)

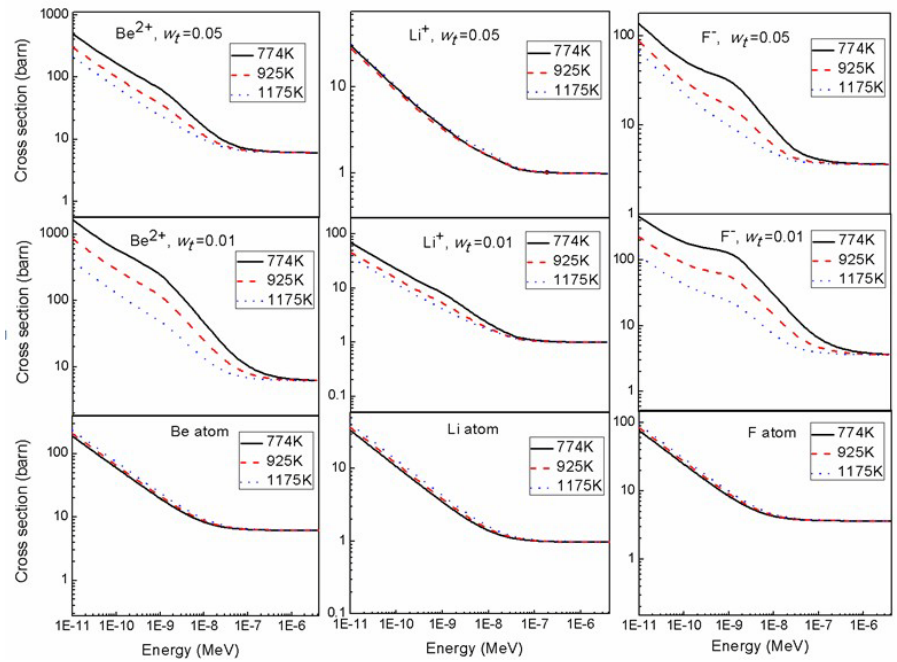
↓

Neutron scattering behavior

↓

Evaluating cross section in different material state

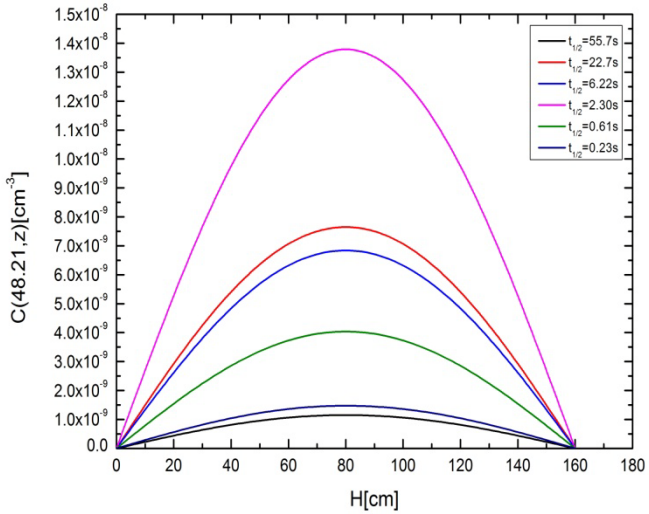
The cross section of liquid FLiBe

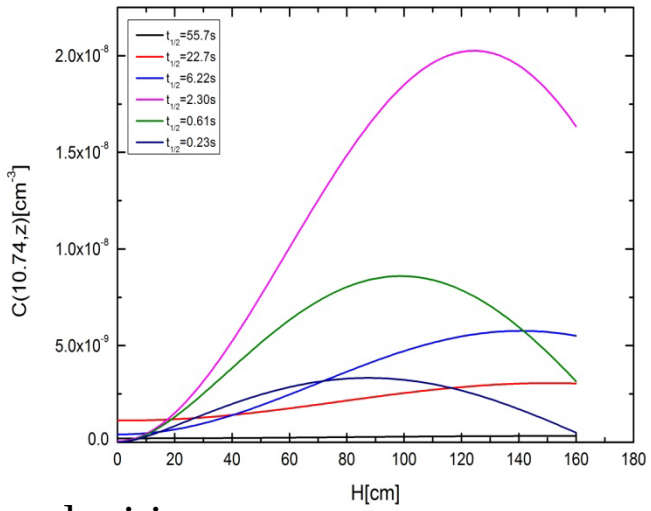


Delay neutron

- Circulating of the fuel salt in TMSR-LF1 will affect β_{eff}
- βeff is an important safety parameter in transient analysis.

Precursors distribution without/with circulating





β_{eff} with different flow velocities.

velocity of flow, cm/s	β_{eff}
22.93	333.6
10.00	453.4
0	640.5

Photoneutron

- Photoneutron in Be contained reactor has significant amount.
- βeff is an important safety parameter in transient analysis, which should contain Photoneutron effect.
- Be is in the salt (TMSR-SF1, TMSR-LF1)

  reaction: Be (γ, n+2α) threshold: 1.67MeV
- Delayed photoneutrons

Delayed photoneutrons in infinite Be per U-235 fission

Group ,i	ϒ-emitter	Energy of ϒ-radiation, MeV	Lifetime	Decay constant,sec^{-1}	Yield $\beta_i(10^{-5})$
1	Kr^{90},Se^{87}	1.78	0.51min	$2.26\cdot10^{-2}$	2.07
2	I^{136},Br^{87}	2.49	1.3 min	$8.85\cdot10^{-3}$	3.66
3	Sb^{133},Kr^{89},Rb^{90}	2.66	3.2 min	$3.60\cdot10^{-3}$	1.85
4	Xe^{138},Rb^{89},Mo^{101}	2.5	15.5 h	$7.42\cdot10^{-4}$	3.68
5	Te^{134},I^{134},Br^{84},Cs^{138}	2.42	43.2 h	$2.67\cdot10^{-4}$	0.36
6	I^{135},La^{142},Kr^{87},Kr^{88}	2.15	3.11 h	$6.20\cdot10^{-5}$	3.20
7	Te^{131}	2.06	21.1 h	$1.59\cdot10^{-5}$	0.260
8	Te^{132}	2.02	77.7 h	$2.48\cdot10^{-6}$	0.038
9	La^{140}	2.51	12.8 days	$6.24\cdot10^{-7}$	0.057

Conclusion

- There are inherent relationships between TMSR-SF1 and TMSR-LF1 on technologies, difficulties, time schedule, etc.
- In TMSR-SF1 and TMSR-LF1 design, safety, feasibility, flexibility and functions are all considered to achieve an engineering and actual reactor.
- Many key physical issues have been studied in TMSR-SF and TMSR-LF to achieve clear comprehensions.
- Before building, there are also many work should be done, such as pebble bed experiments, thermal neutron scattering cross section of FLiBe, a more reliable crossing section database, engineering uncertainties analysis, benchmark, etc.

THANK YOU

FOR YOUR ATTENTION