Molten Salt Reactor Transients

How Power of The Future Performs In Accidents

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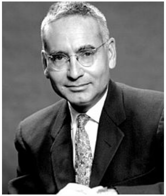
Outline

Energy for the future

Cheap and abundant nuclear energy is no longer a luxury; it will eventually be a necessity for the maintenance of the human condition. – Alvin Weinberg



Molten salt reactors offer a convincing solution to the problem of fossil fuels.

- Potentially much cheaper than normal nuclear
- Make meltdowns impossible
- Better natural resource utilization

MSR Comparison

Topaz Solar Farm

- 9.5 sq. mi of California desert
- Max output of 550 MW(e), on average makes 132 MW(e)
- $2.5B construction

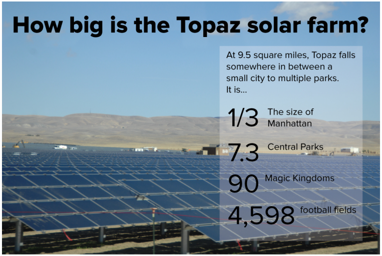


Figure 1: Topaz solar farm in California, credit GigaOM media

Terrestrial Energy IMSR concept

- About 300 MW(e) output, more than double Topaz farm
- Initial cost estimates rank IMSR cheaper than coal

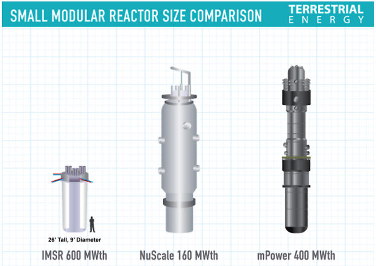


Figure 2: IMSR and some other small nuclear designs, credit Terrestrial Energy

Benchmark Case: MSRE

- Constructed at Oak Ridge National Lab, ran reliably 1965-1969 at 7.4 MW(th)
- Various tests proved theory and tech-readiness for full-scale power plants
- Stopped operation since all needed experiments were done

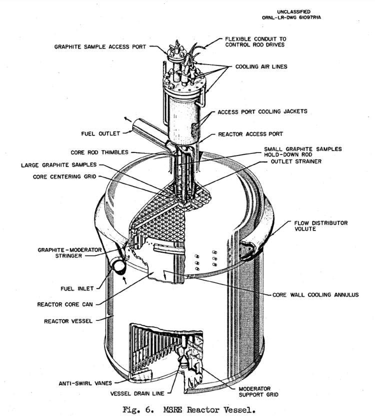
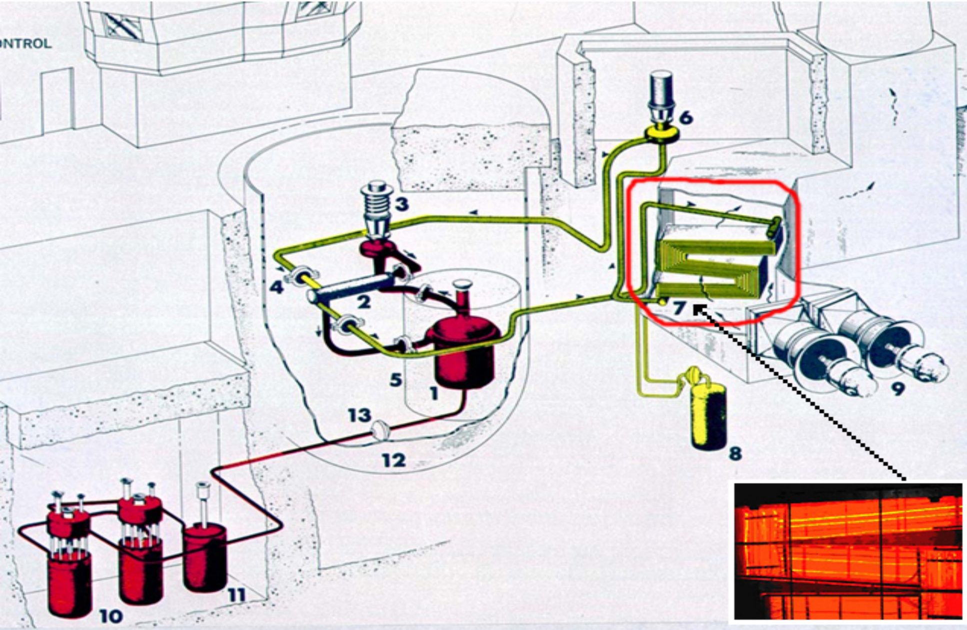


Figure 3: MSRE reactor core diagram from ORNL technical reports



1. Reactor Vessel, 2. Heat Exchanger, 3. Fuel Pump, 4. Freeze Flange, 5. Thermal Shield,

MSRs: An Intrinsically Coupled System I

Neutrons' changing concentrations can be described approximately with coupled diffusion equations:

$$\frac{1}{v_g}\frac{\partial \phi_g}{\partial t} = \nabla \cdot D_g \nabla \phi_g + \sum_{g \neq g'}^{G} \Sigma^s_{g' \to g}\phi_{g'} + \chi^p_g \sum_{g'=1}^{G}(1-\beta)\nu\Sigma^f_{g'}\phi_{g'} + \chi^d_g \sum_{i}^{I}\lambda_i C_i - \Sigma^r_g \phi_g \tag{1}$$

where v_g = speed of neutrons in group g
ϕ_g = flux of neutrons in group g
t = time
D_g = Diffusion coefficient for neutrons in group g
Σ^r_g = macroscopic cross-section for removal of neutrons from group g
$\Sigma^s_{g' \to g}$ = macroscopic cross-section of scattering from g' to g
χ^p_g = prompt fission spectrum, neutrons in group g
G = number of discrete groups, g
ν = number of neutrons produced per fission
Σ^f_g = macroscopic cross section for fission due to neutrons in group g
χ^d_g = delayed fission spectrum, neutrons in group g
I = number of delayed neutron precursor groups
β = delayed neutron fraction
λ_i = average decay constant of delayed neutron precursors in precursor group i

MSRs: An Intrinsically Coupled System II

Delayed neutron precursors are products of freshly split uranium that emit a new neutron after a delay. *critical* to reactor control; they shift power change timescales from picoseconds to seconds despite only being a few tenths of a percent of emitted neutrons.

$$\frac{\partial C_i}{\partial t} = \sum_{g'=1}^{G} \beta_i \nu \Sigma_{g'}^{f} \phi_{g'} - \lambda_i C_i - \frac{\partial}{\partial z} u C_i \tag{2}$$

Heat and temperature affect the coefficients in Equation 1 significantly. Energy conservation must be solved:

$$\rho_f c_{p,f} \frac{\partial T_f}{\partial t} + \nabla \cdot (\rho_f c_{p,f} \vec{u} \cdot T_f - k_f \nabla T_f) = Q_f \tag{3}$$

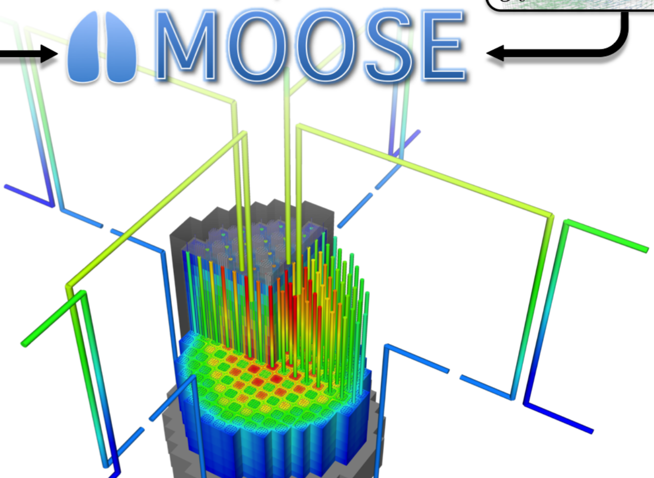
ρ_f = density of fuel salt
$c_{p,f}$ = specific heat capacity of fuel salt
T_f = temperature of fuel salt
$\vec{u}$ = velocity of fuel salt
k_f = thermal conductivity of fuel salt
Q_f = source term

Outline

$$\vec{\Omega}\cdot\vec{\nabla}\Psi+\sigma_t(\vec{r})\Psi(\vec{r},\vec{\Omega}) = \frac{1}{4\pi}(\sigma_s(\vec{r})\Phi(\vec{r})+S(\vec{r}))$$

$$\nabla\cdot k\nabla T = 0$$

$$\frac{\partial c}{\partial t} - \nabla\cdot(\vec{v}c) = 0$$



MOOSE Physics Representation

- Highly object-oriented code solves weak form of PDE using finite element method
- PETSc solves resulting system of nonlinear equations using generalized minimal residual method (GMRES)
- Some of the world's most cutting-edge numerical algorithms and scalable parallel computing are made painlessly accessible to the everyday user

MOOSE Example

In MOOSE, the term $D\nabla^2 u$ is easily represented by:

```
Real
GroupDiffusion::computeQpResidual()
{
  return _D[_qp][_group] * _grad_test[_i][_qp] *
         computeConcentrationGradient(_u, _grad_u, _qp);
}
```

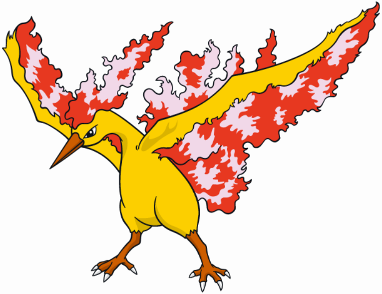
A vacuum boundary condition in neutronics calculations can easily be represented by:

```
Real
VacuumConcBC::computeQpResidual()
{
  return _test[_i][_qp] * computeConcentration(_u, _qp) / 2.;
}
```

Moltres

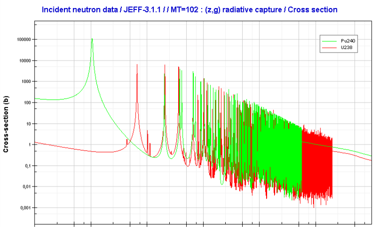
- github.com/arfc/moltres
- Publicly developed on github
- Continuous integration by Civet
- Includes detailed guide for contributing

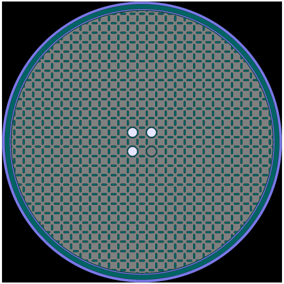
Most importantly: physics kernels, boundary conditions, and even capability to couple into MOOSE Navier-Stokes solvers provided to the moltres user

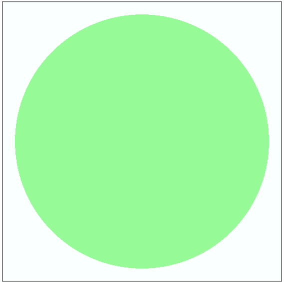


Group constant generation

- Where do coefficients like Σ, D, χ, ν come from?
- **Serpent 2 - Monte Carlo neutronics**
  - Continuous energy, meaning very close to first-principles physics
  - Trace millions of neutrons through reactor geometry
- *Behemothly* computationally expensive for transients, accuracy scales $\mathcal{O}(\sqrt{n}^{-1})$

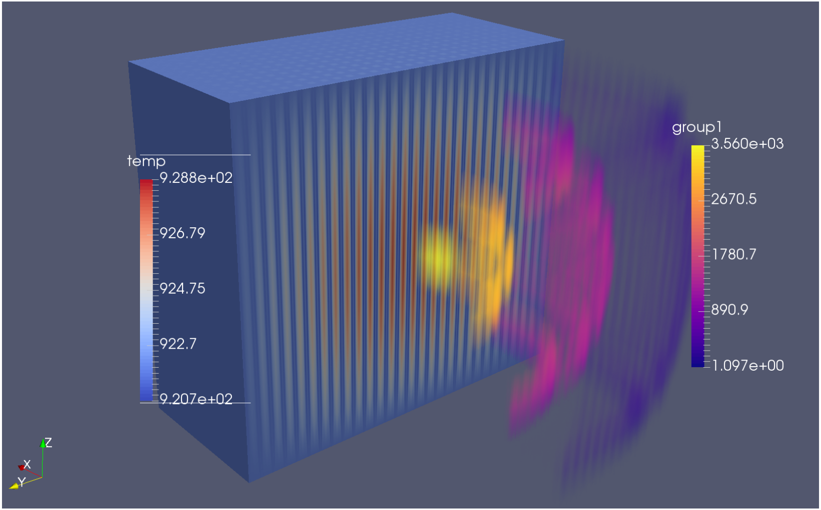




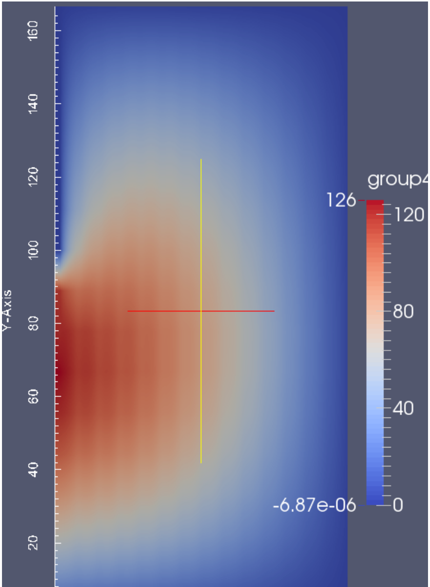


Outline

Results


temp
9.288e+02
926.79
924.75
922.7
9.207e+02
group1
3.560e+03
2670.5
1780.7
890.9
1.097e+00
Z
X
Y

More Results



Conclusion

Moltres can contribute to building power for the future.

- Group constants generated for MSRE-like reactors, made public on github
- Work in progress to automatically model salt heat exchangers, salt loops

Future work

- Will build small research reactors, transport effects may dominate
- $\implies$ implement P_3 or so transport
- Include molten salt-specific thermalhydraulics models

Acknowledgement

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