Activities for Safety Assessment of Fast Spectrum Systems

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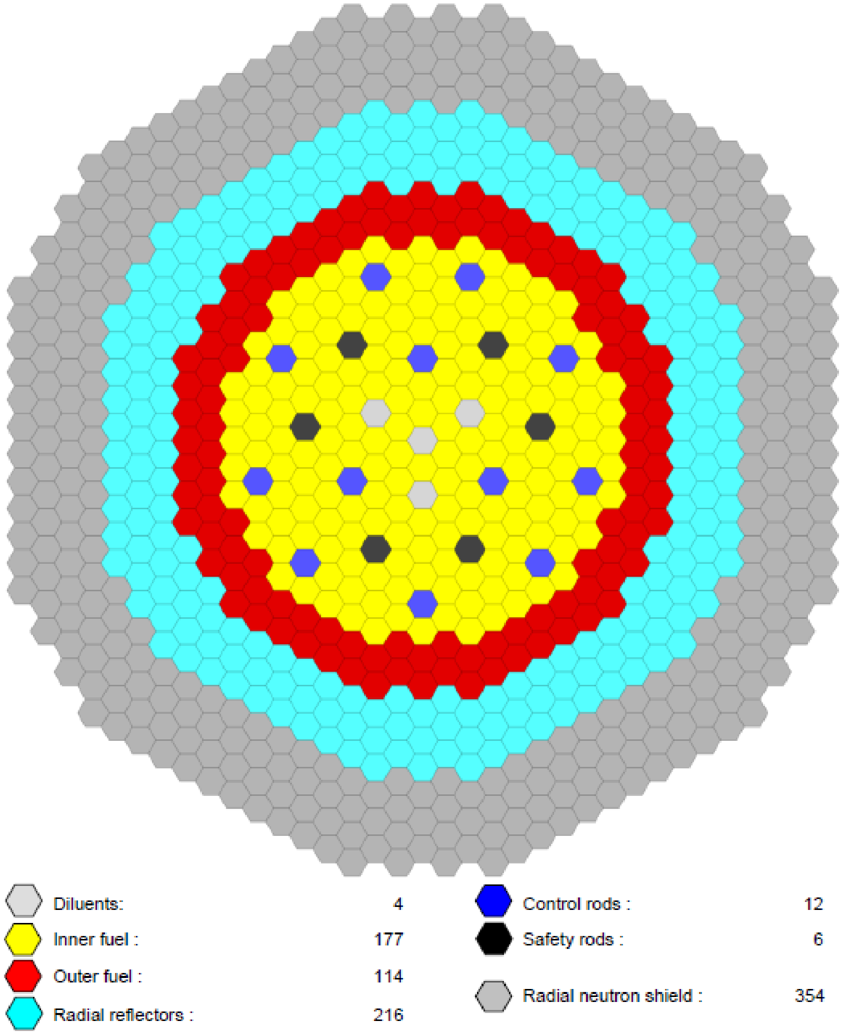
Content

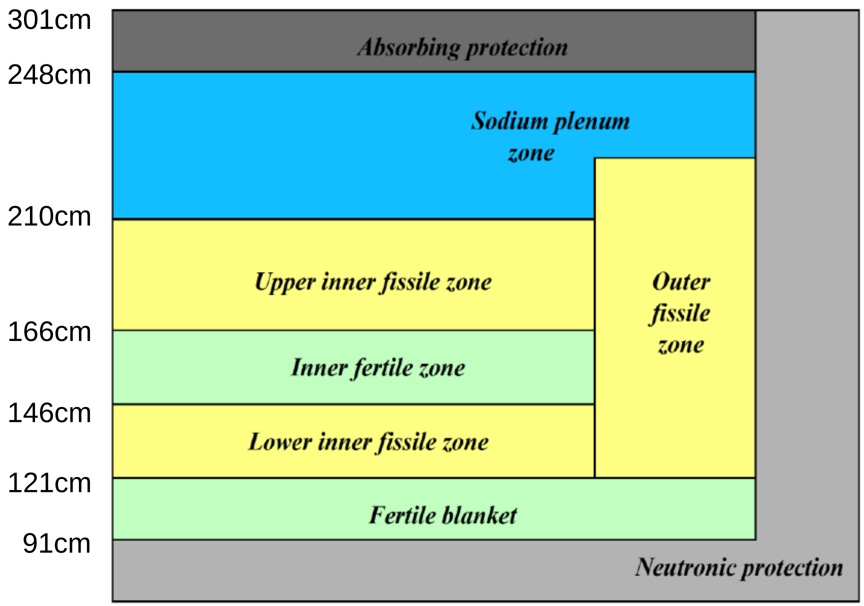
- R&D for the safety assessment of liquid metal cooled fast spectrum systems
  - Thermal hydraulics of liquid metals (ATHLET)
  - Simulation of time-dependent distributed neutron sources (PARCS)
  - Modeling of radial and axial core thermal expansion (FEM/PARCS)
  - Simulation of spallation neutron sources (MCNPX)
  - MAXSIMA (MYRRHA)
  - ESNII+ (ASTRID) neutronic and thermal-hydraulic simulation

Generic ASTRID Core Design

Radial core layout:



Axial core layout:



Flat-to-flat assembly pitch at nominal operation: 17,611 cm

ASTRID Neutronic Modeling with HELIOS

- HELIOS 1.12
  - 190 energy group library, unadjusted
  - 112 energy group library

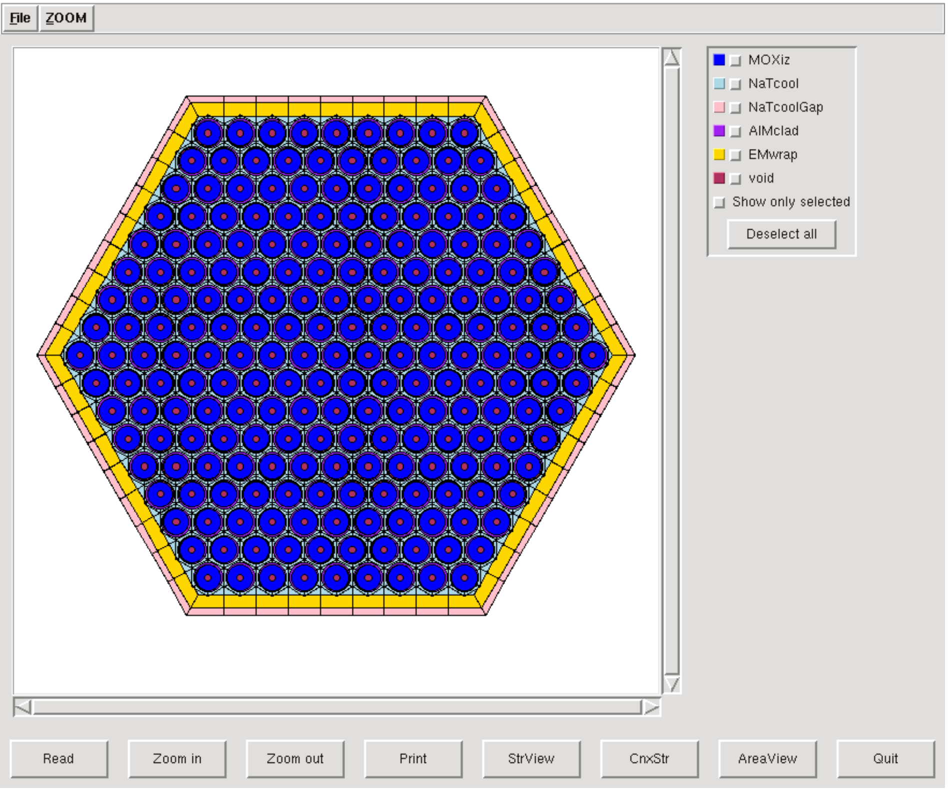
- 8 energy group structure:

Energy group index	Lower limit (eV)	Energy group index	Lower limit (eV)
1	2.2313E+6	5	4.0868E+4
2	8.2085E+5	6	1.5034E+4
3	3.0197E+5	7	7.4852E+2
4	1.1109E+5	8	1.0000E-4

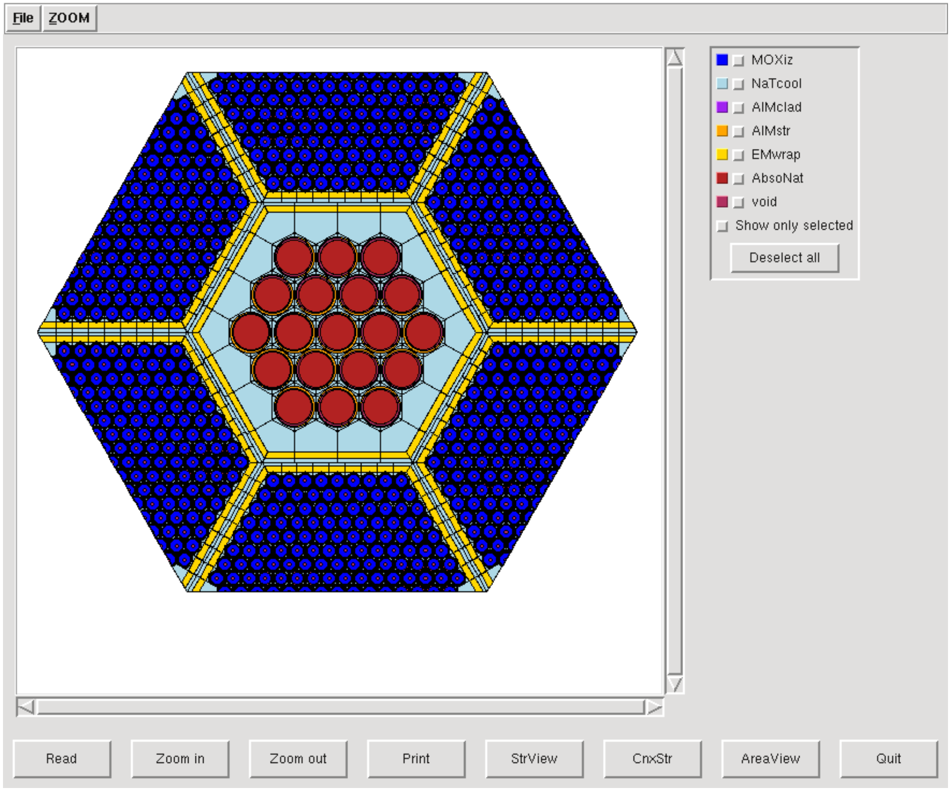
- For selected comparative test calculations
  - Serpent v2.1.21
  - JEFF-3.1 nuclear data

HELIOS Models (examples)

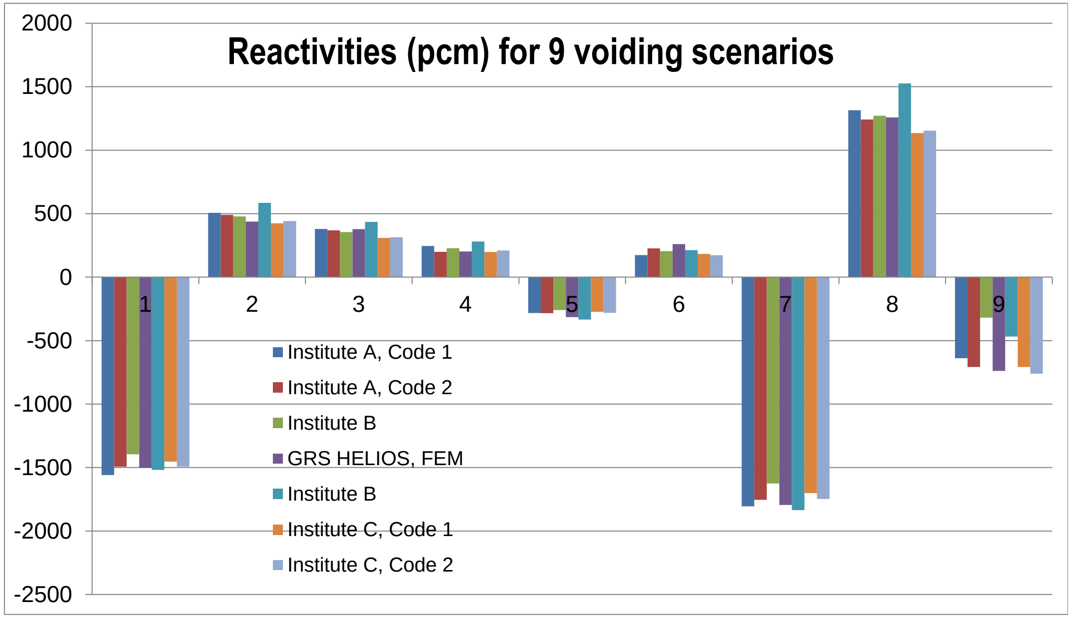
- Fissile/fertile assemblies:

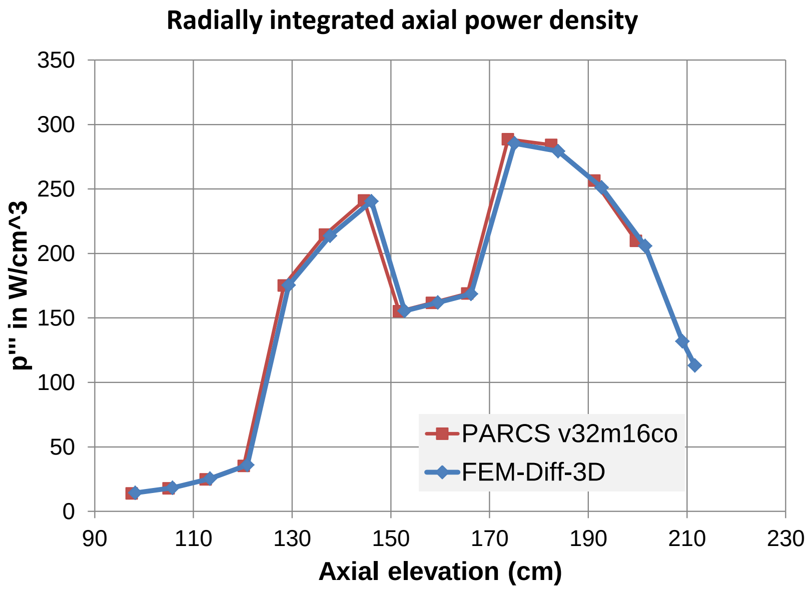


CSD/DSD assemblies:



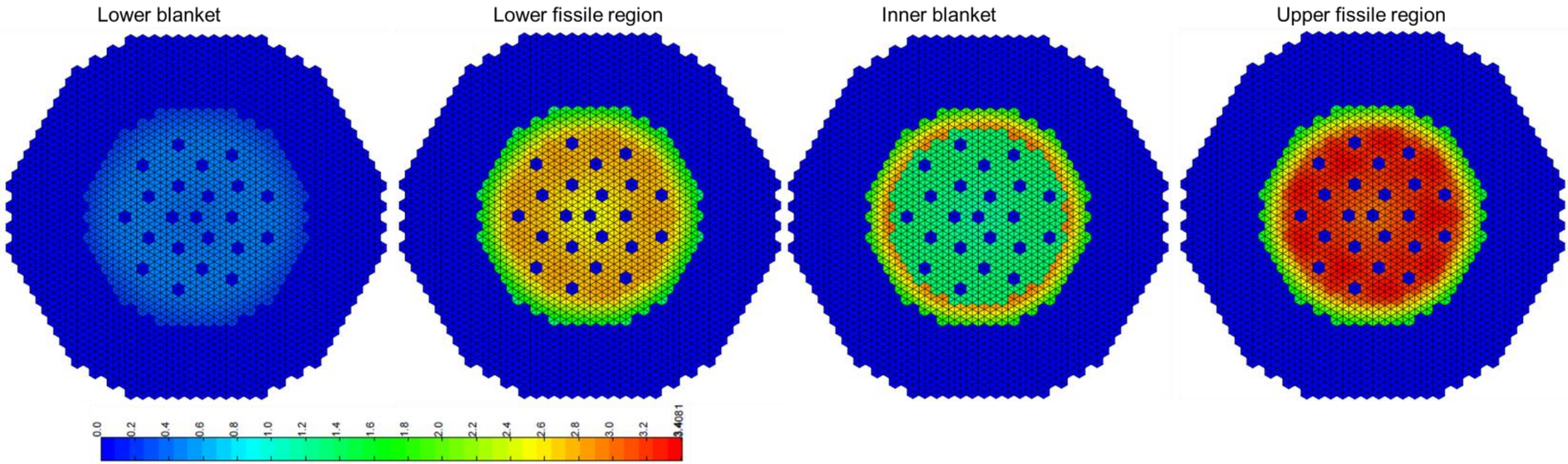
Evaluation of Core Safety Parameters (HELIOS/PARCS/FEM)





Evaluation of Core Safety Parameters

- Radial power distributions



ASTRID Thermal-Hydraulic Modeling with ATHLET 3.0b

Three single SAs

- Inner hot SA `CORE-IH`
- Inner average SA `CORE-I1`
- Outer SA `CORE-O1`

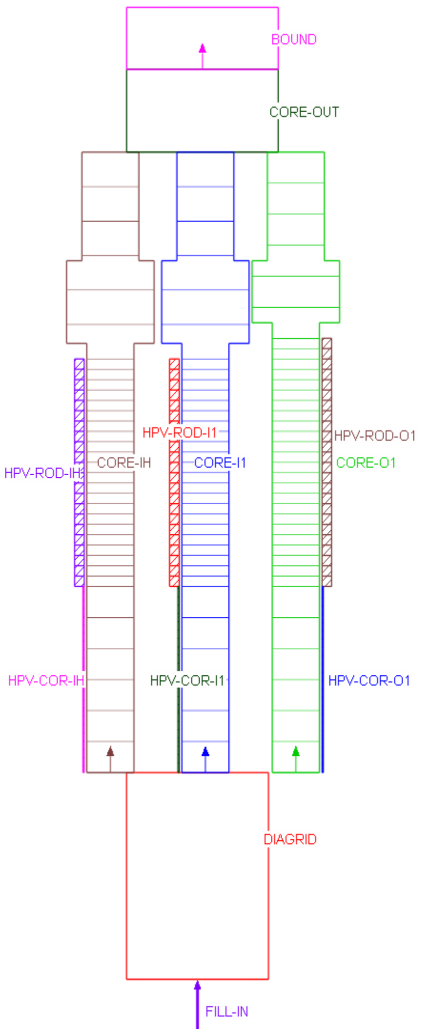
Mass flow controlled by common fill `FILL-IN` entering `DIAGRID`

Mass flow distribution in SAs according to individual flow loss coefficients

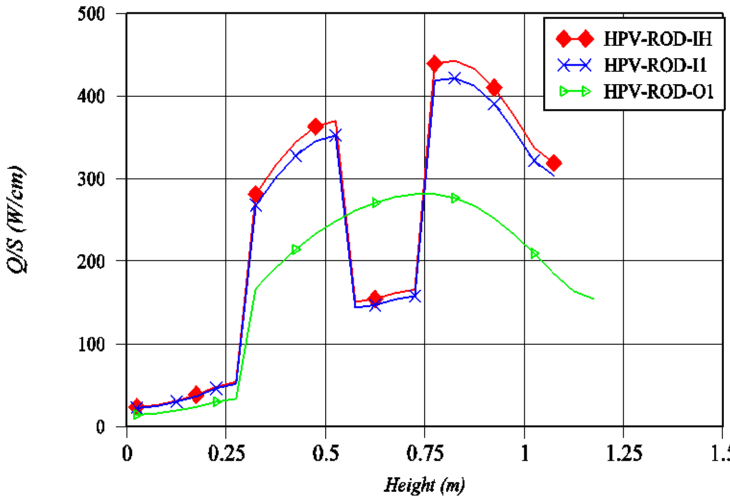
Pressure controlled by time dependent volume `BOUND`

SAs

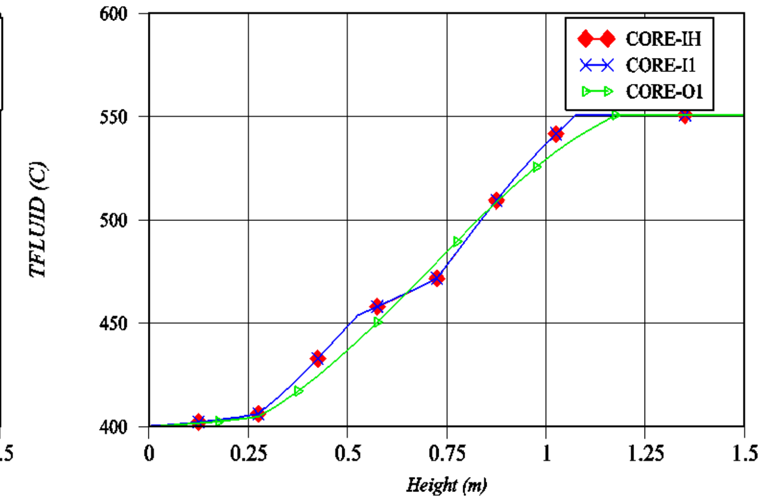
- Geometric data, mass flows, diagrid porosity etc. according to ESNII+ specifications
- One representative heated rod group for each SA pipe
- One unheated rod group for each SA pipe in lower section (`-IH`, `-I1`, `-O1`) (only structure of cladding)



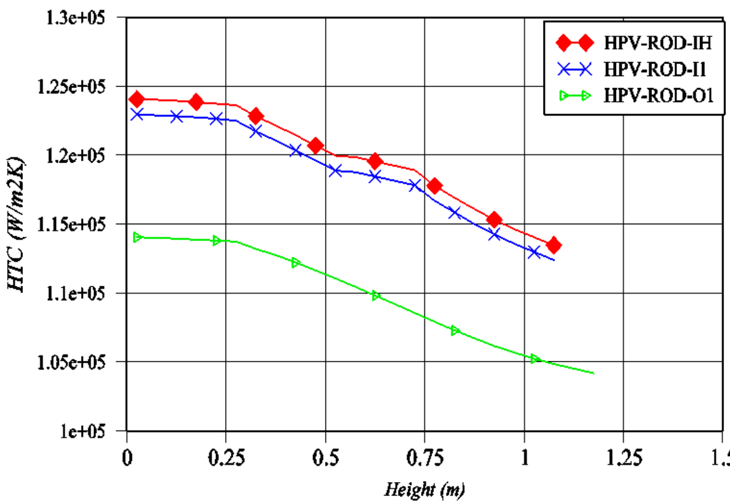
ASTRID Steady State TH Simulation with ATHLET 3.0b



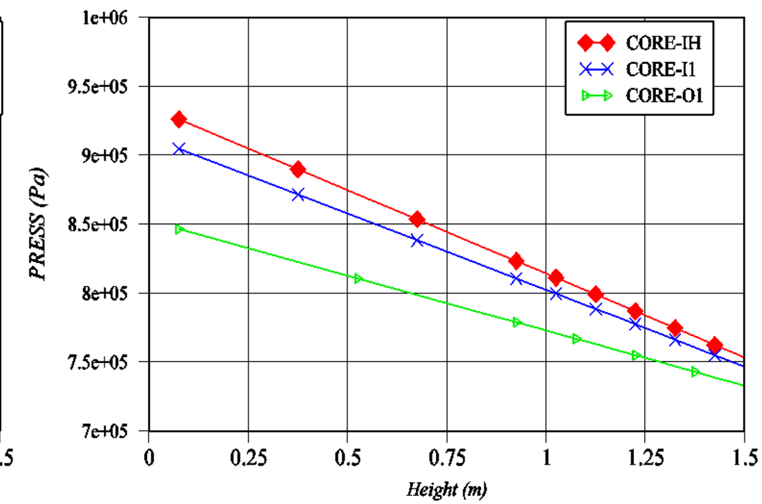
Linear heat generation rate



Fluid Temperature in core channels



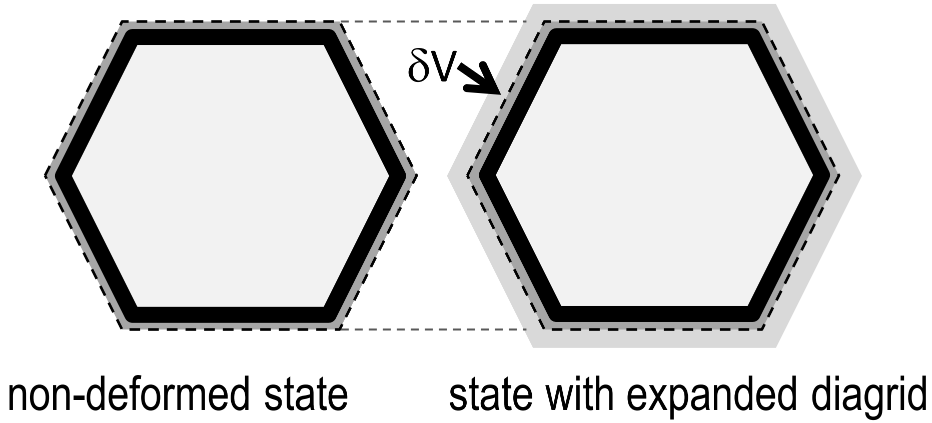
HTC to fluid



Pressure core channels

Diagrid Thermal Expansion Modeling

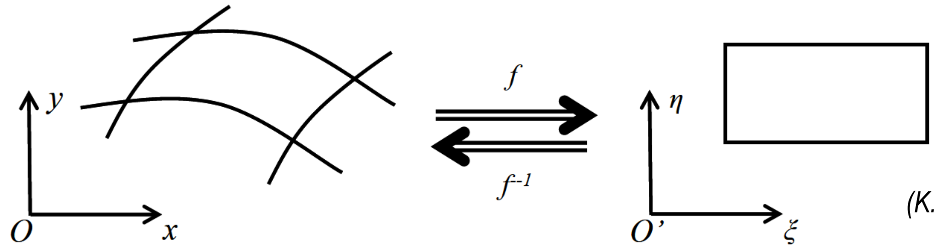
- In an SFR, increasing coolant inlet temperature causes thermal expansion of the diagrid plate, i.e. the assembly core support structure.



- Enlarged spacing between adjacent subassembly wrappers – spacing is filled by coolant.
- Associated reactivity changes may present a large (negative) contribution to the total reactivity feedback.
- Assembly pitch changes affect the radial spatial meshing of the core simulator.
- Aim: Treat pitch changes with the core simulator's fixed radial meshing.

Diagrid Thermal Expansion Modeling

- Mapping the meshing of the radially deformed core (x, y, z) to the meshing of the non-deformed core (ξ, η, z):



(K. Azekura, T. Hayase, J. Nucl. Sci. Techn. **26** (1989) 374)

- Diffusion equation of the non-deformed core: $-D_g(\vec{r})\Delta\Phi_g(\vec{r}) + \sigma_g^r(\vec{r})\Phi_g(\vec{r}) = q_g(\vec{r})$
- Approximate diffusion equation of the deformed core:

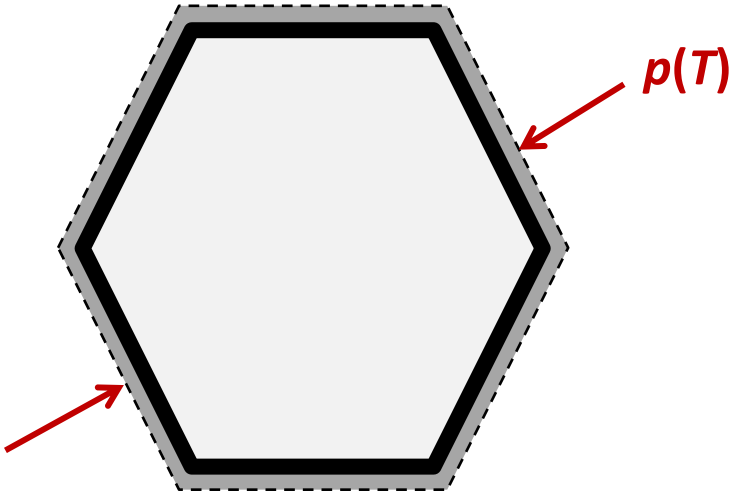
$$-\frac{D'_g(\vec{r})}{\left(\frac{\partial x}{\partial \xi}\right)^2}\cdot\frac{\partial^2\Phi'_g}{\partial\xi^2} - \frac{D'_g(\vec{r})}{\left(\frac{\partial y}{\partial \eta}\right)^2}\cdot\frac{\partial^2\Phi'_g}{\partial\eta^2} + \sigma'^r_g(\vec{r})\Phi'_g(\vec{r}) = q'_g(\vec{r})$$

$$q'_g(\vec{r}) = \sum_{g'\neq g}\sigma'^s_{gg'}(\vec{r})\Phi'_{g'}(\vec{r}) + \frac{\chi_g}{k'}\sum_{g'\neq g}\sigma'^f_{g'}(\vec{r})\Phi'_{g'}(\vec{r})$$

Different multiplication factor

- Change of cross sections to account for enlarged inter-assembly gap

Diagrid Thermal Expansion Modeling



Thermal expansion correlation for diagrid

- Subassembly pitch thermal expansion:

$$p(T) = p(20°C) \cdot \left(1 + \varepsilon_{SS316}(T)\right)$$

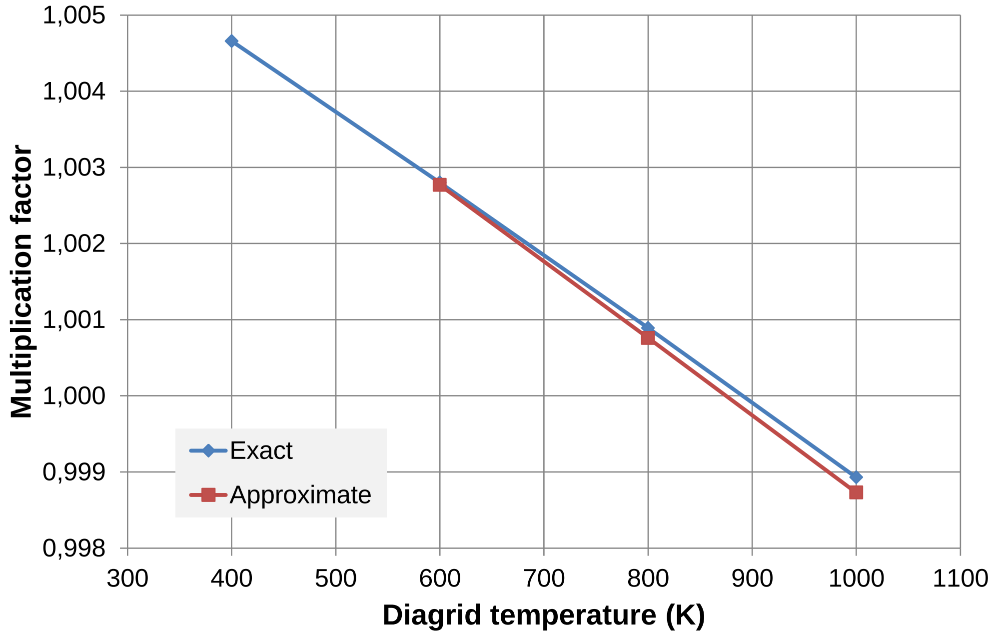
- Thermal expansion of SS316

$$\varepsilon_{SS316}(T) = a(T - 20°C) + b(T - 20°C)^2 + c(T - 20°C)^3$$

with coefficients a, b, c given within ESNII+ project WP6

Results for Diagrid Expansion of the ASTRID Generic Design

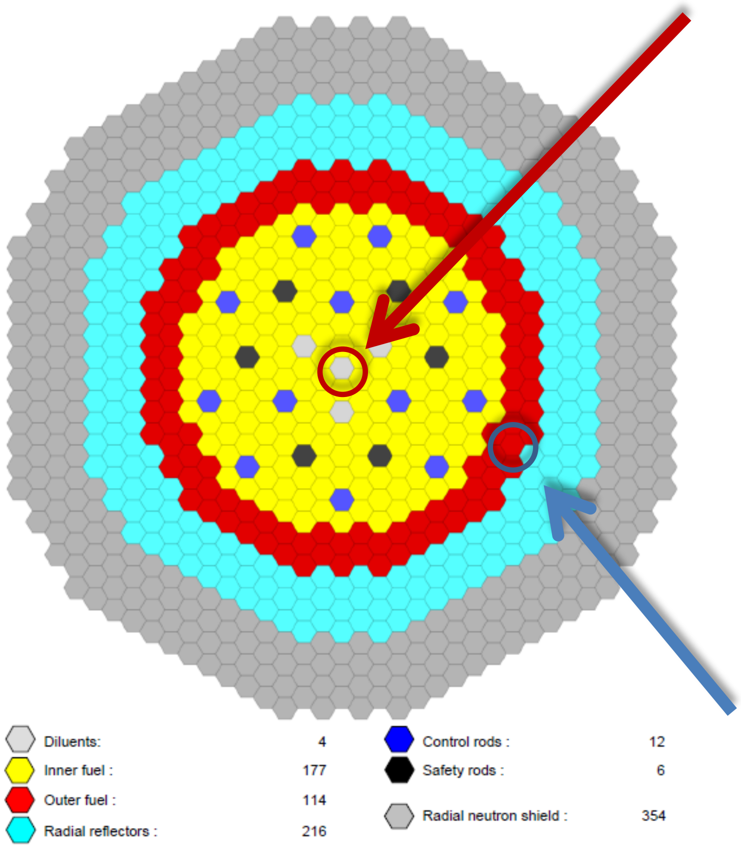
- Implementation: Fortran module
- Made available first to an FEM few-group diffusion code
- Application to ASTRID generic design within ESNII+ project
- Ref.: *A. Seubert et al., Annual Meeting on Nuclear Technology, Berlin, Germany, 2015*
- Could be also implemented in PARCS in future

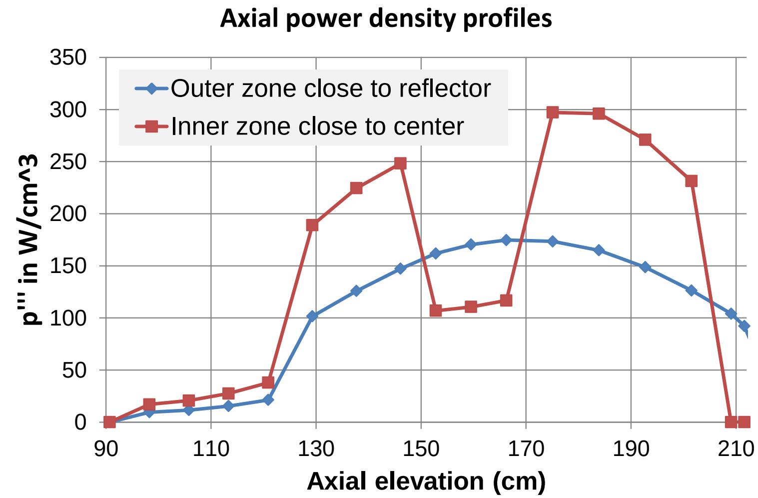


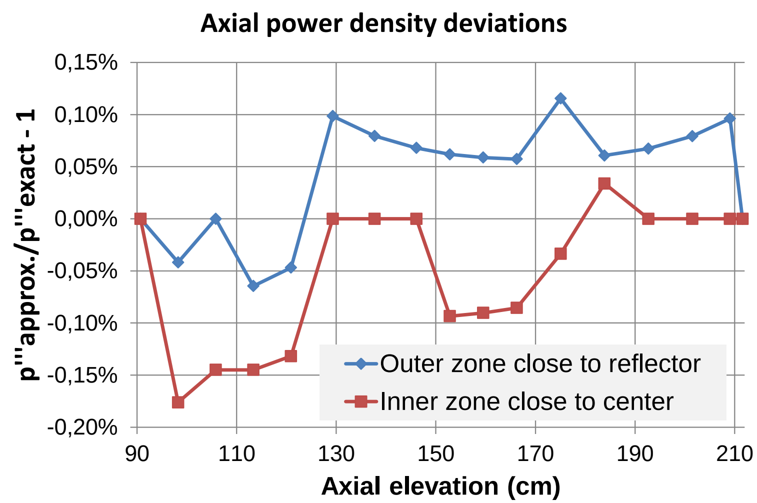
Diagid temperature	Flat-to-flat pitch (cm)	k_{eff}^{exact}	Reactivity change $\Delta\rho$ (pcm)	k_{eff}^{approx}	Deviation from exact $\Delta\rho$ (pcm)
400 K (nom.)	17.611	1.00466	–	–	–
600 K	17.674 (+0.36%)	1.00280	–185	1.00277	–3
800 K	17.741 (+0.74%)	1.00089	–375	1.00076	–13
1000 K	17.809 (+1.1%)	0.99893	–570	0.99873	–20

Results for Diagrid Expansion of the ASTRID Generic Design

Axial power density profiles
in individual subassemblies

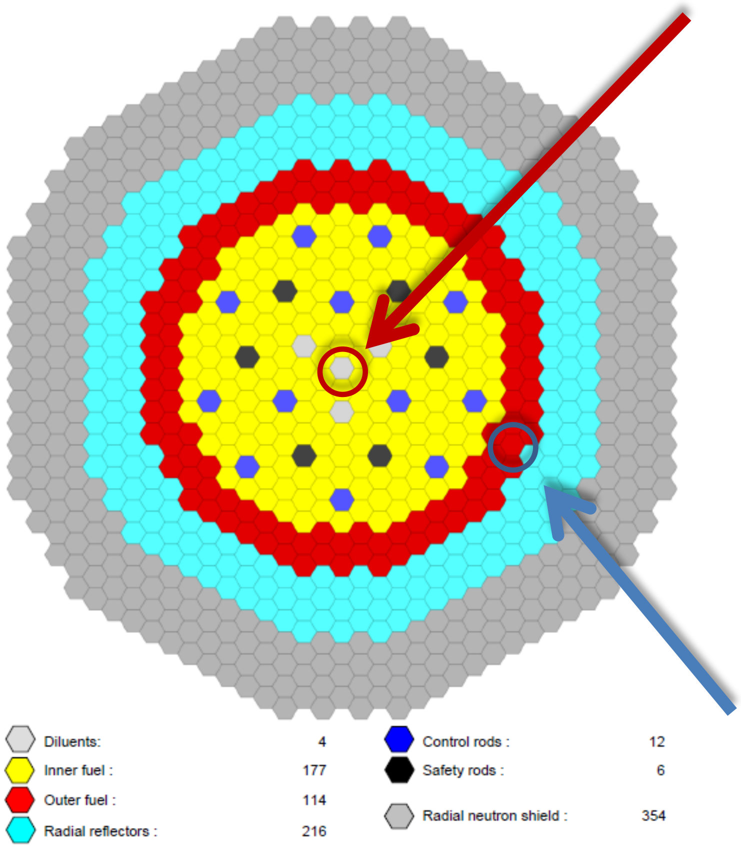


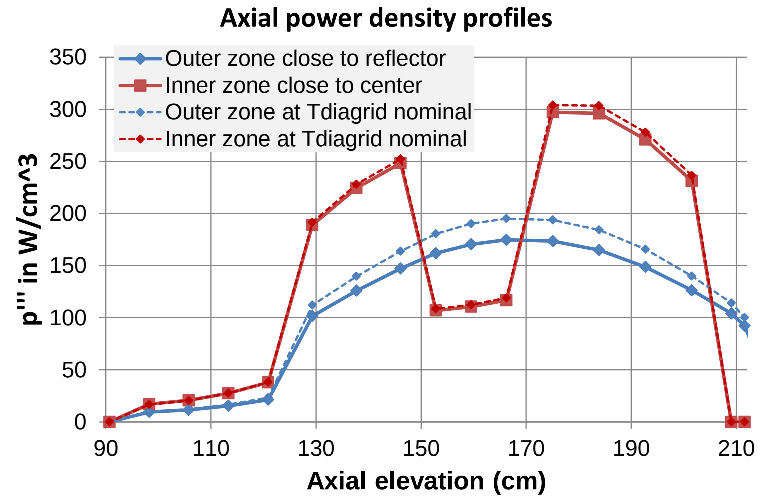


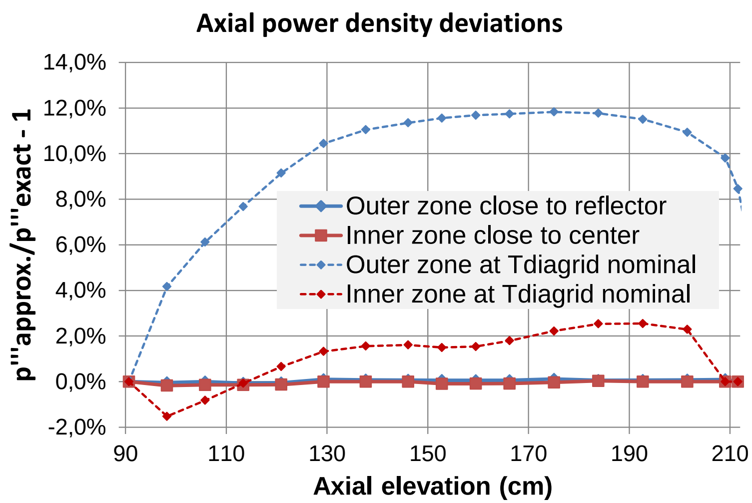


Results for Diagrid Expansion of the ASTRID Generic Design

Axial power density profiles in individual subassemblies compared with nominal diagrid temperature

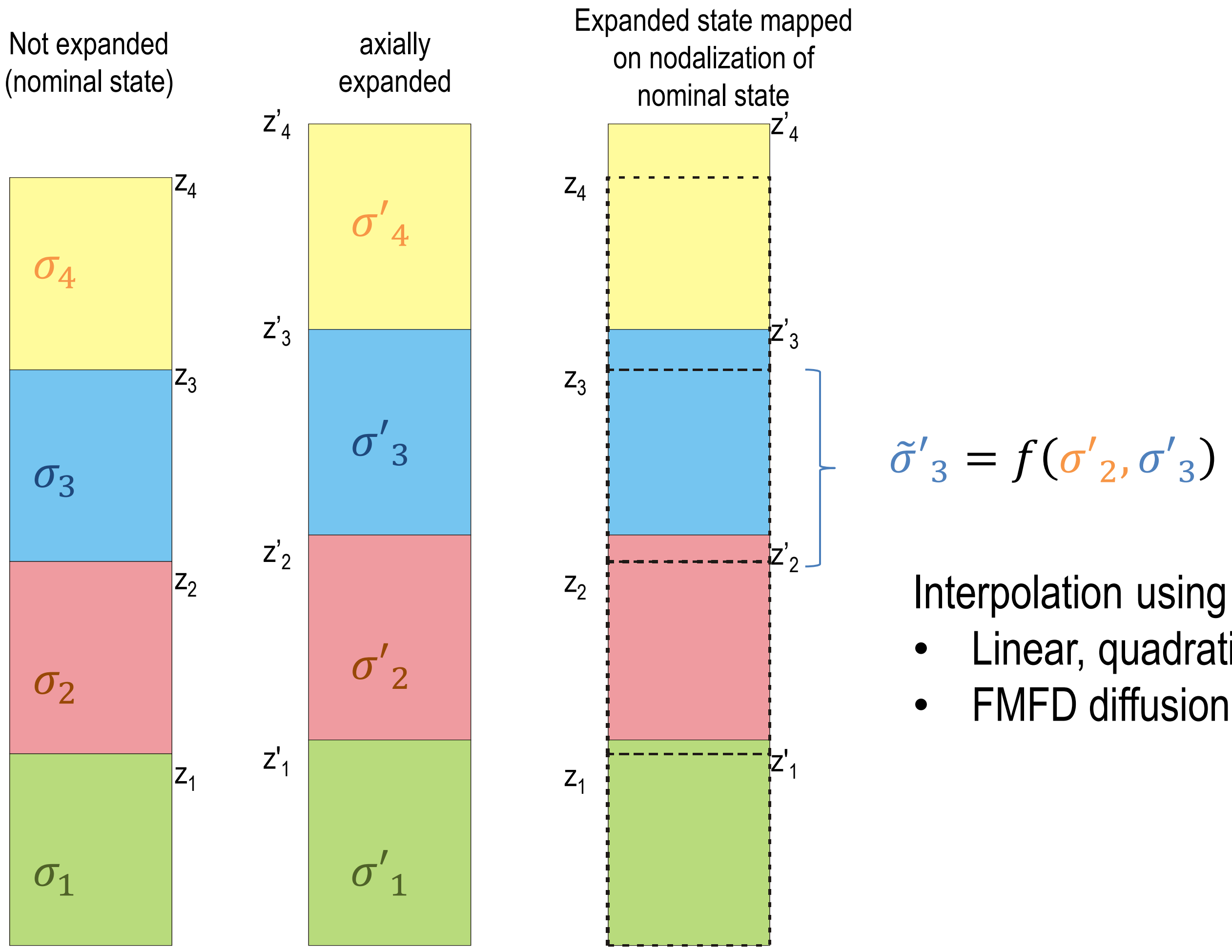






Axial fuel/cladding expansion modeling

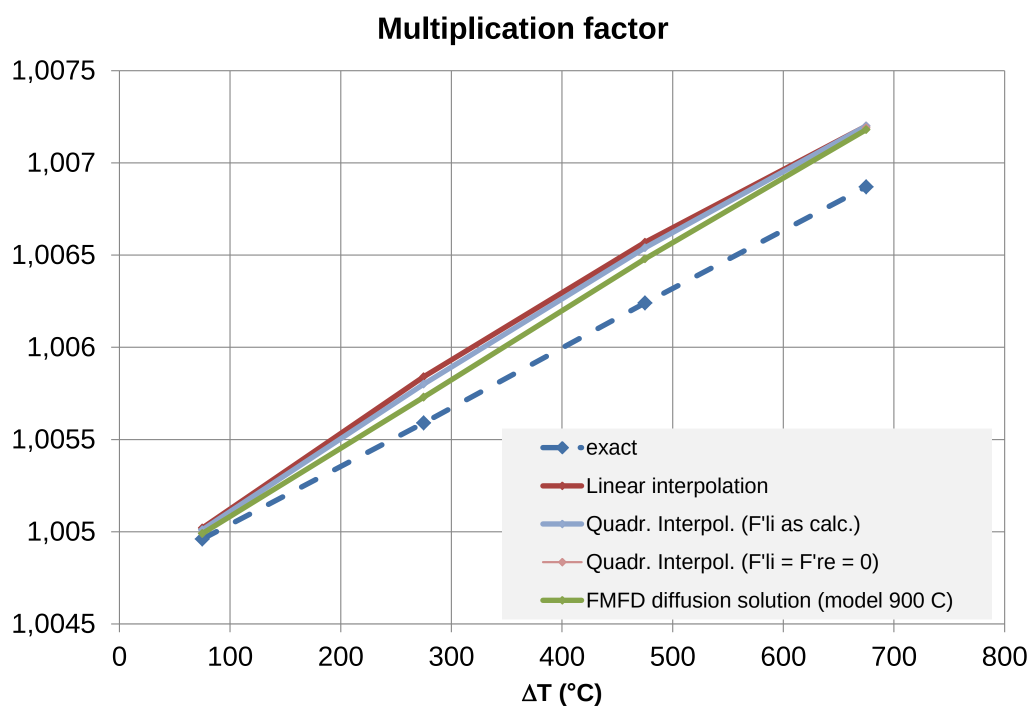
- Problem: Keep axial nodalization of the core simulator unchanged
- Assembly pitch unchanged, simultaneous radial pin expansion treated by cross section library parameter



Interpolation using either:
- Linear, quadratic, cubic neutron flux
- FMFD diffusion neutron flux solution

Axial fuel/cladding expansion modeling

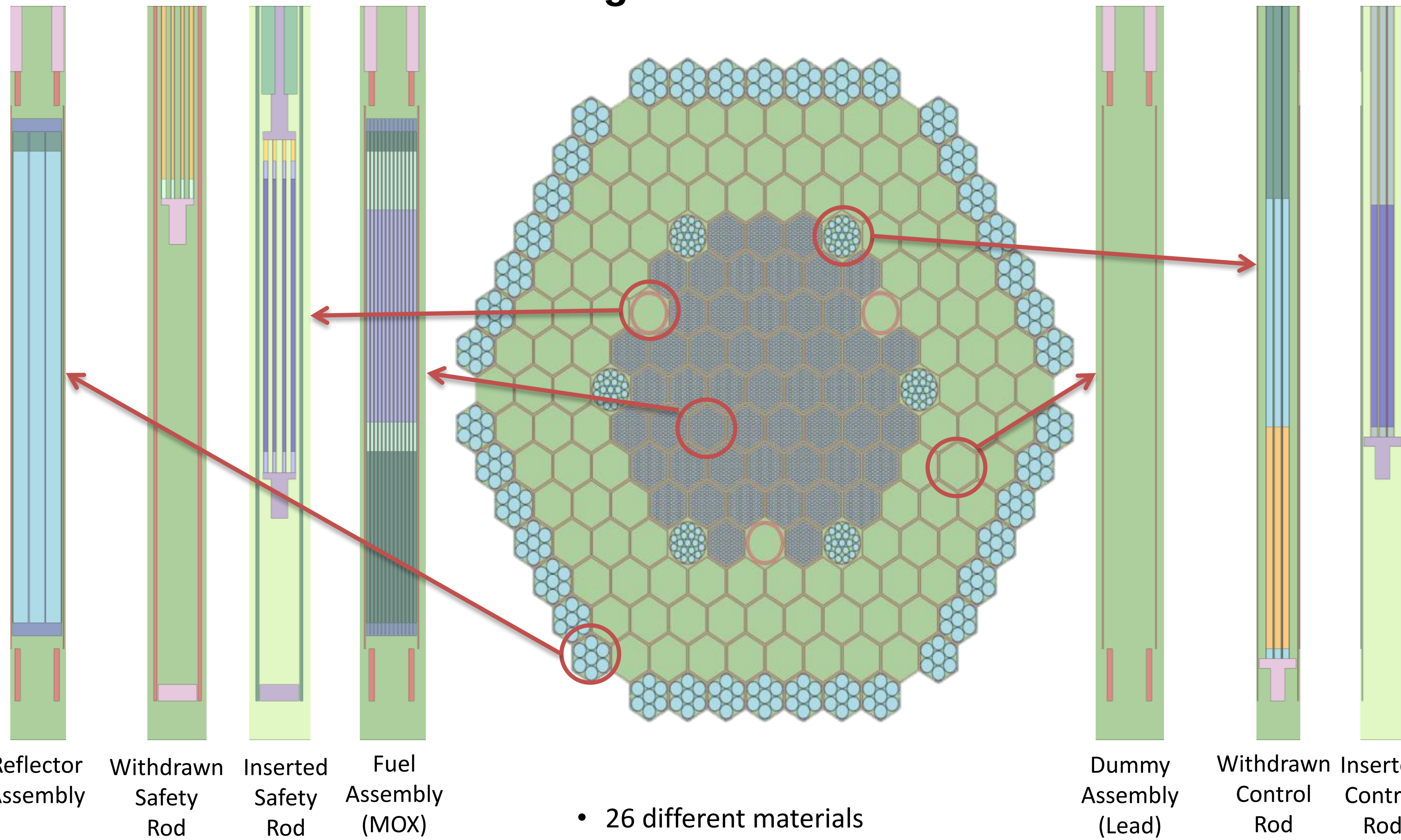
- Test calculations of axial expansion modeling for ASTRID
- Nuclear material properties fixed – consistent changes in nuclide density and radial pin dimensions will be treated by a few-group nuclear cross section library parameter
- Linear thermal expansion according to ESNII+ specifications

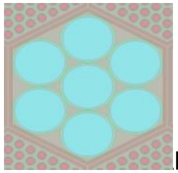
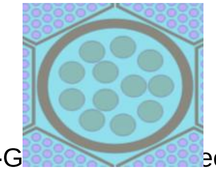
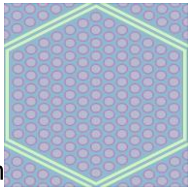


ΔT/°C	exact		
	k	Δρ	Δρ/ΔT
75	1,00496		
275	1,00559	62	0,312
475	1,00624	127	0,321
675	1,00687	189	0,311

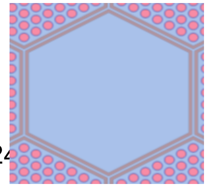
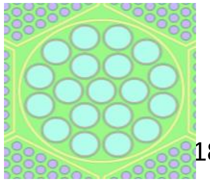
FMFD diffusion solution (model 900 C) delta_z = 1,0 cm, nz_min = 2, nz_max = 4	
k	Δρ w.r.t. exact
1,00499	3
1,00573	14
1,00648	24
1,00718	31

MYRRHA Critical Core Modeling with SERPENT and PARCS



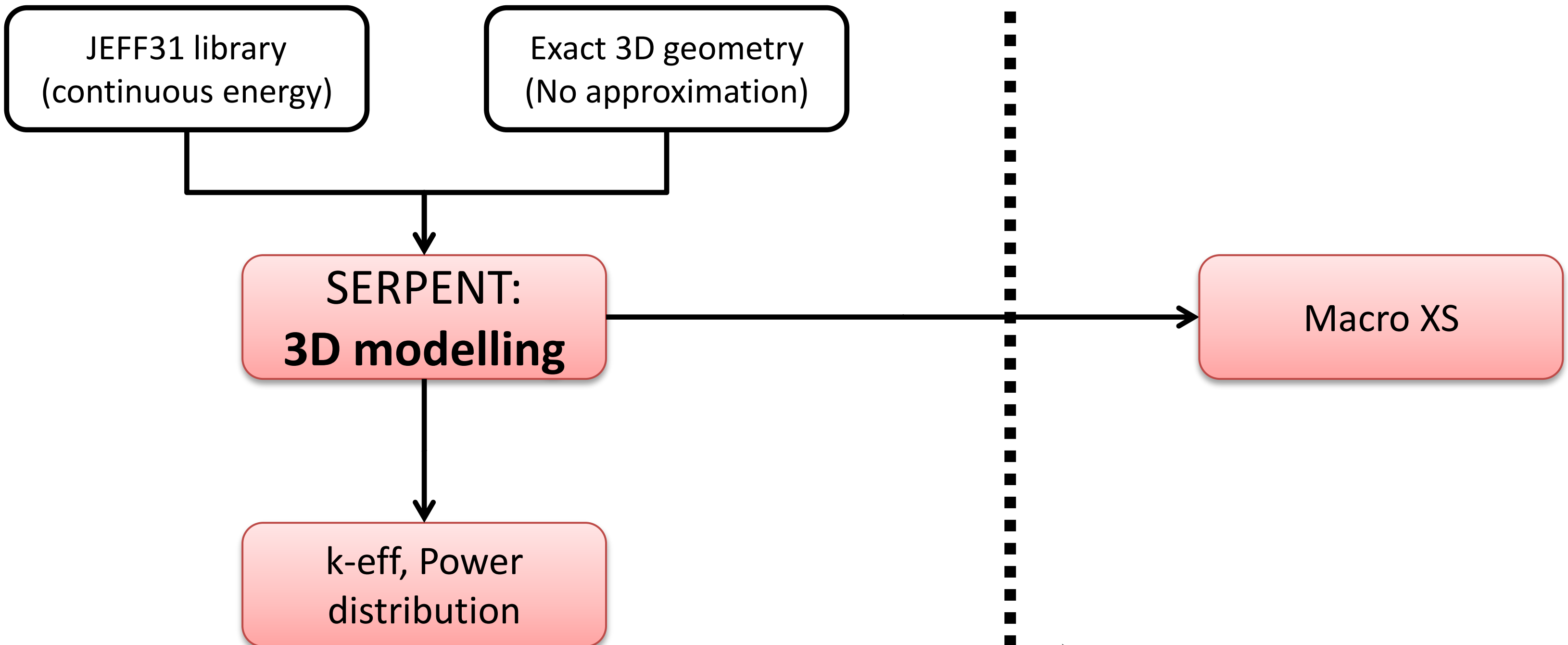




- 26 different materials
- Hexagonal geometry
- Complex geometry for safety and control rod

Reference calculation: Monte-Carlo calculation

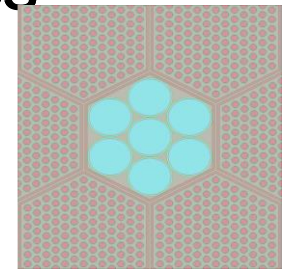
Preparation of XS for PARCS calculations

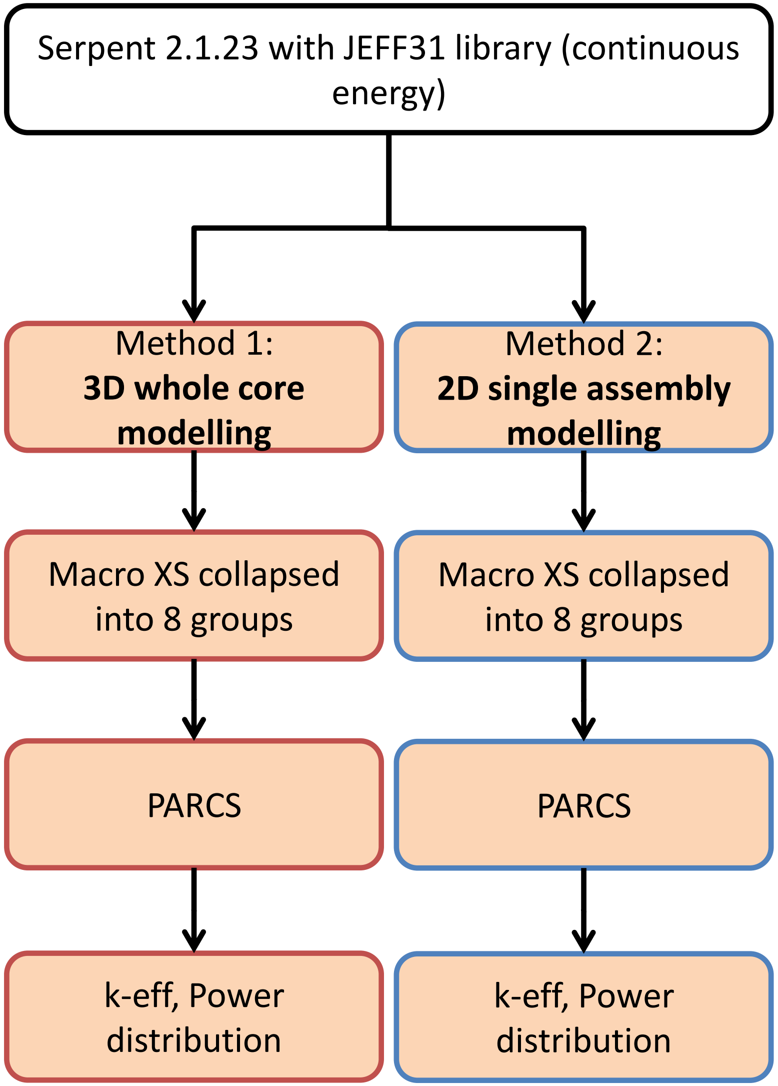


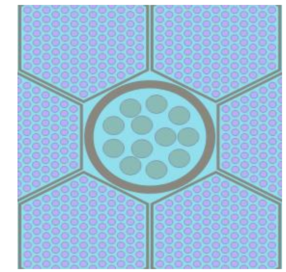
- Number of source neutrons per cycle: 200 000
- Number of active cycles run: 2000
- Number of inactive cycles run: 200

- Calculated with the 3D neutron flux
- Collapsed into 8 energy groups for every type of material

2 methods to generate few-group cross sections for PARCS

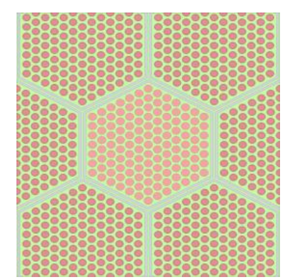






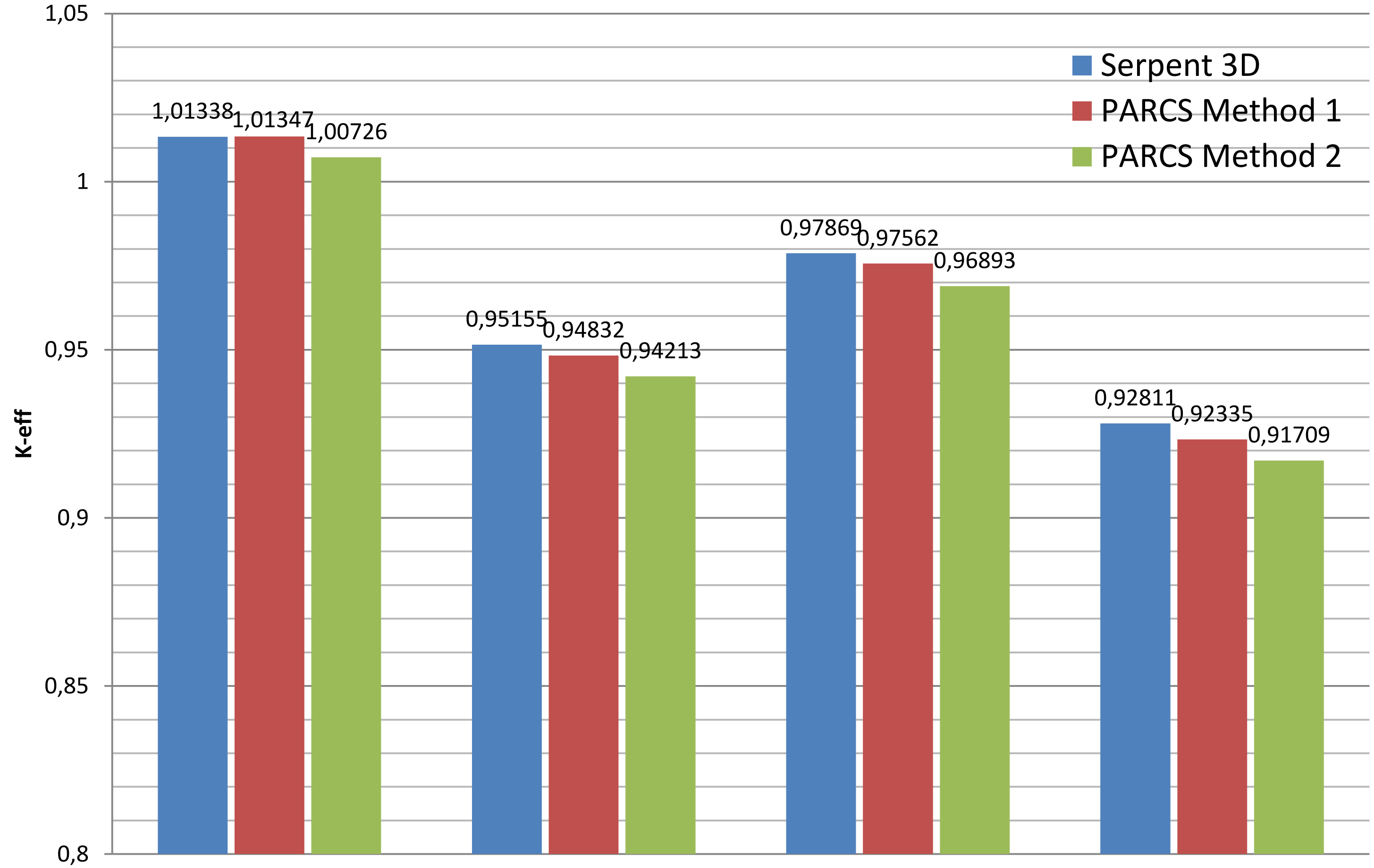
- Realistic 3D geometry
- 3D neutron flux

- infinite lattice
- periodic boundary condition



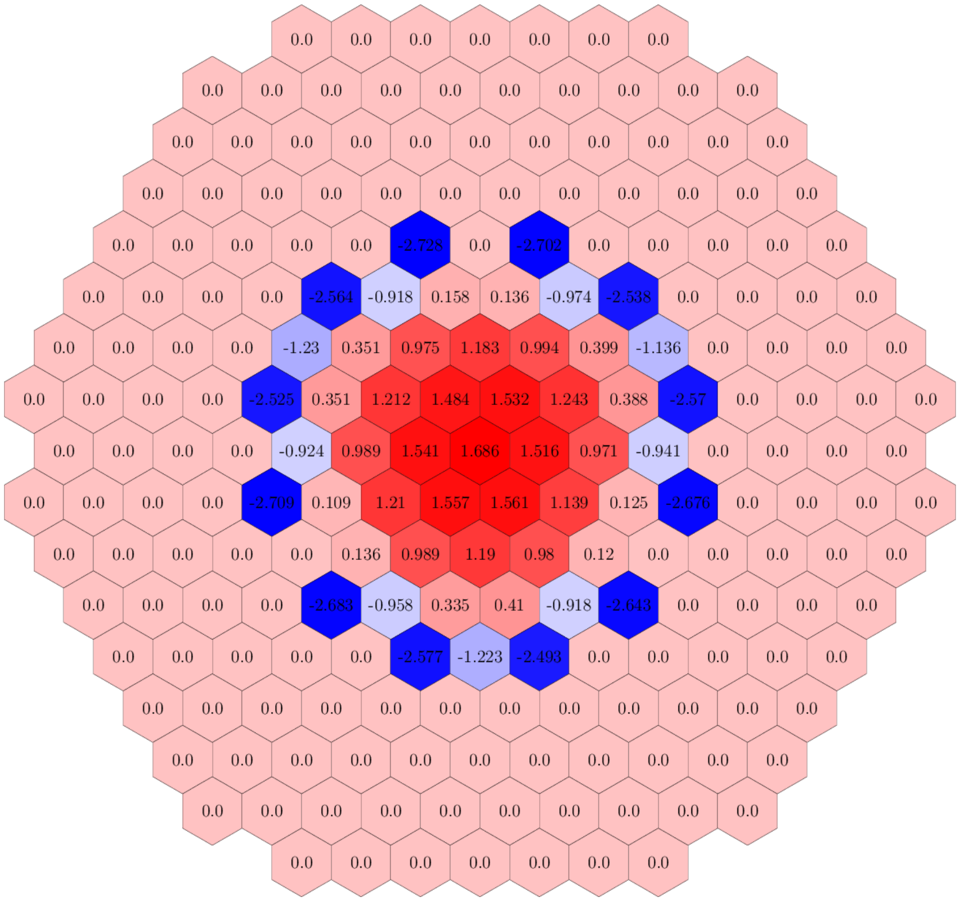
8 energy groups structure Lower energy limits (eV)	
2.23E+06	4.09E+04
8.21E+05	1.50E+04
3.02E+05	7.49E+02
1.11E+05	1.00E-04

Multiplication factor results (preliminary)

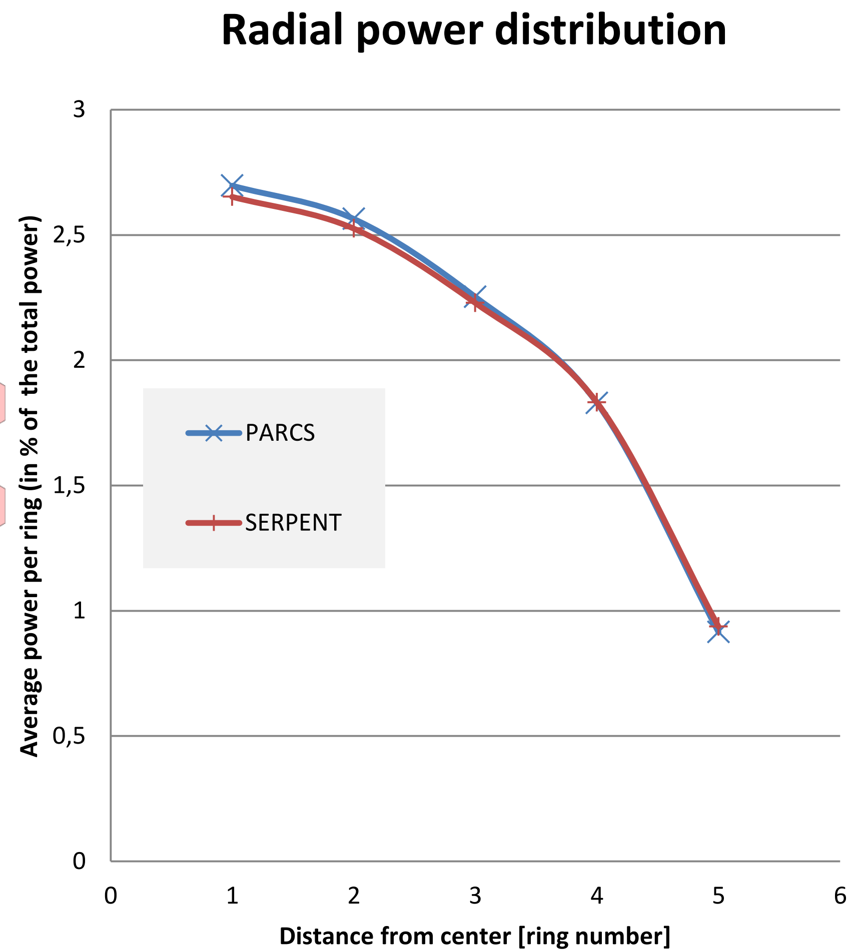


Control Rods position	Withdrawn	Inserted	Withdrawn	Inserted
Safety Rods position	Withdrawn	Withdrawn	Inserted	Inserted

Radial power distribution comparison (preliminary)



Relative difference between PARCS (Method 1) and Serpent results for each FA (in %)



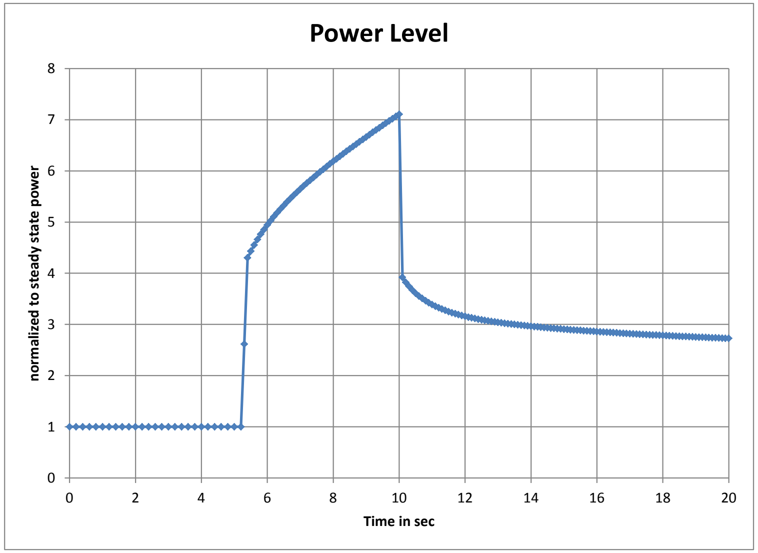
Extension of PARCS for External Neutron Source Simulation

- Objective: Simulation of ADS
- Experience from implementation in TORT-TD and validation by YALINA-Thermal
- Transport equation:

$$\left[\frac{1}{v_g}\frac{\partial}{\partial t}+\hat{\vec{\Omega}}\cdot\vec{\nabla}+\sigma_g^{tot}(\vec{r})\right]\psi_g(\vec{r},\vec{\Omega},t)=q_g(\vec{r},\vec{\Omega},t)+\sum_{g'}\int_{4\pi}d\vec{\Omega}'\sigma_{gg'}(\vec{r},\vec{\Omega}'\cdot\vec{\Omega})\psi_{g'}(\vec{r},\vec{\Omega}',t)+$$

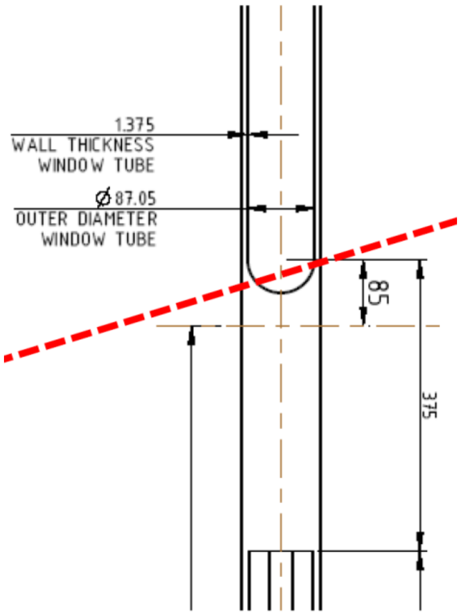
$$+\chi_g(1-\beta)\sum_{g'}\upsilon\sigma_{fg'}(\vec{r})\phi_{g'}(\vec{r},t)+\sum_l \chi_{gl}^d\lambda_l c_l(\vec{r},t)$$

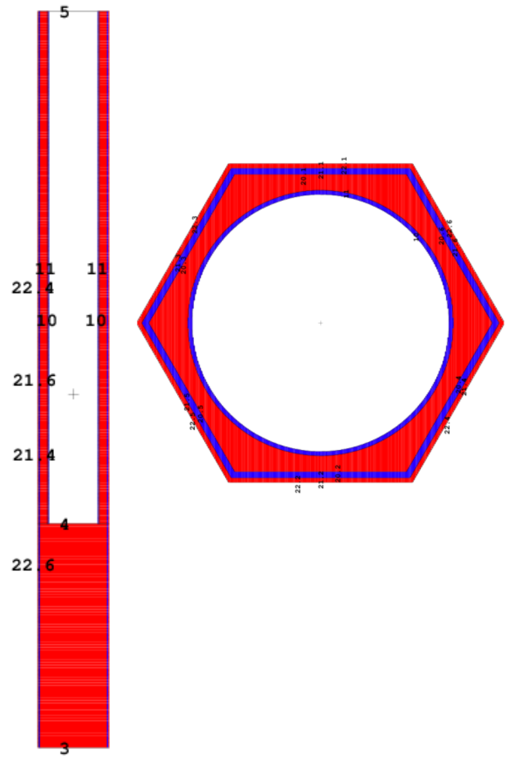
- Example for a single localized source pulse:

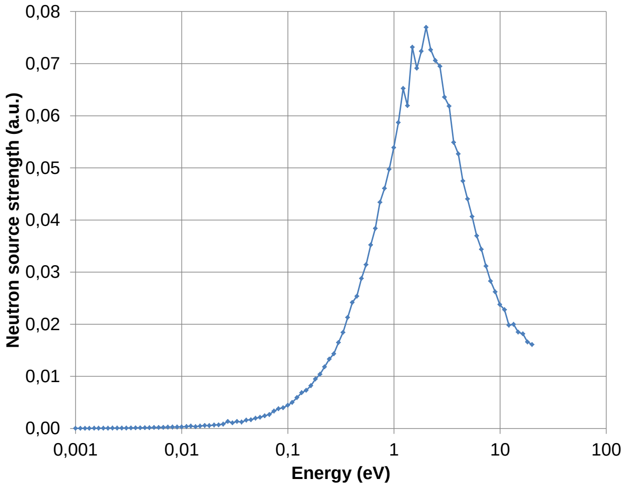


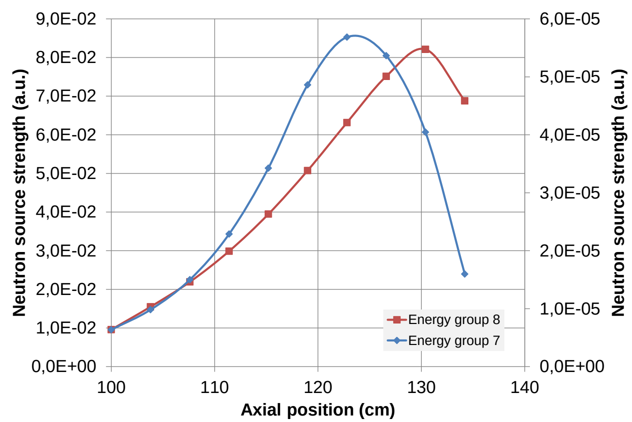
Simulation of Spallation Neutron Sources

- MCNPX model of the MYRRHA spallation target









Summary

Activities at GRS for Safety Assessment of Fast Spectrum Systems

- Systems:
  - Sodium cooled fast reactors (e.g. ASTRID)
  - Liquid metal cooled fast spectrum systems, including source driven sub-critical systems (e.g. MYRRHA)
- Codes:
  - ATHLET, PARCS, HELIOS, SERPENT, MCNPX
- Modeling extensions:
  - Thermal hydraulics of liquid metals (ATHLET)
  - Simulation of time-dependent distributed neutron sources (PARCS)
  - Modeling of radial and axial core thermal expansion
  - Simulation of spallation neutron sources (MCNPX)
- Few-group nuclear cross-section generation (HELIOS, SERPENT)
- Whole core neutronics modeling (PARCS, SERPENT)