Technical Meeting on
Liquid Metal Reactor Concepts: Core Design
and Structural Materials, June 12-14, 2013

Tools and applications for core design and shielding in fast reactors

Presented by: Reuven Rachamin

HZDR

HELMHOLTZ
ZENTRUM DRESDEN
ROSSENDORF

Outline

- Modeling of SFR cores using the Serpent-DYN3D code sequence
- Core shielding assessment for the design of FASTEF-MYRRHA
- Neutron shielding studies on an advanced Molten Salt Fast Reactor (MSFR) design

Modeling of SFR cores using the Serpent-DYN3D code sequence

Reuven Rachamin, Emil Fridman

Introduction

- DYN3D nodal code
  - Developed at HZDR for LWR application
  - 3D full core steady-state and transient calculations
  - Multi-group diffusion and SP3 solvers
  - Square and hexagonal geometries
  - Is being extended for SFR analysis

- Important tasks
  - Selection of appropriate lattice code
  - Establishment of a few-group XS generation procedure

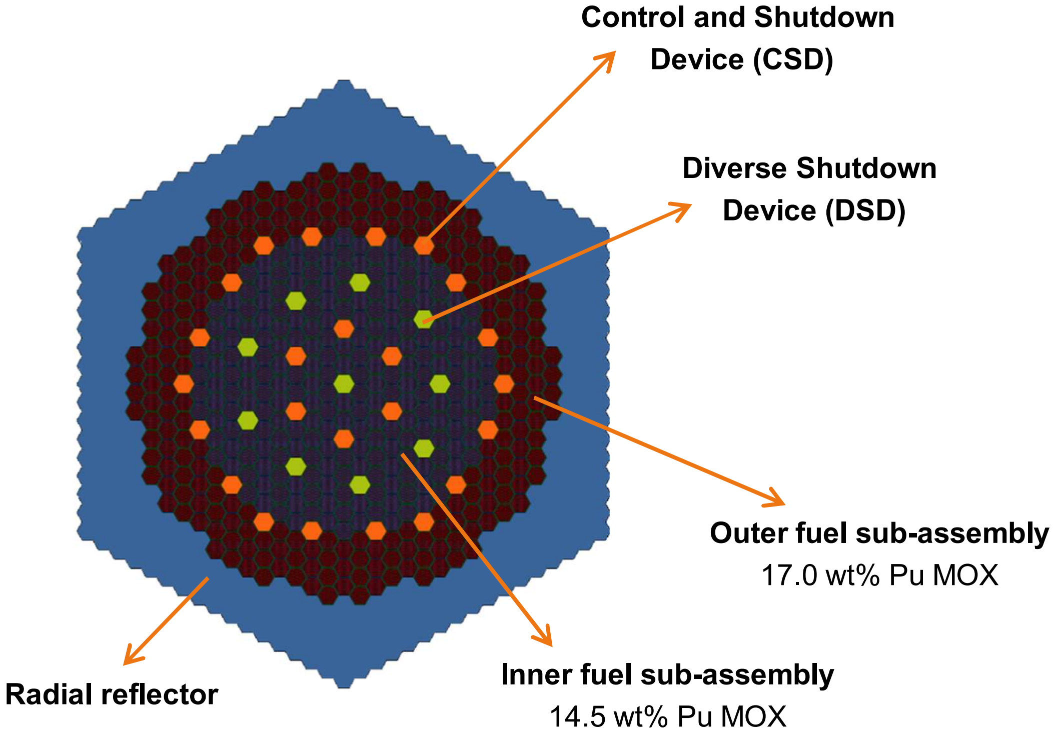
- Candidate lattice codes
  - Serpent Monte Carlo transport code
  - HELIOS deterministic lattice transport code

Objectives

- To establish few-group XS generation procedure
  - For SFR cores analysis with DYN3D
  - Using Serpent
- To investigate the performance of Serpent-DYN3D sequence
  - Via 2D full core modeling of SFR core

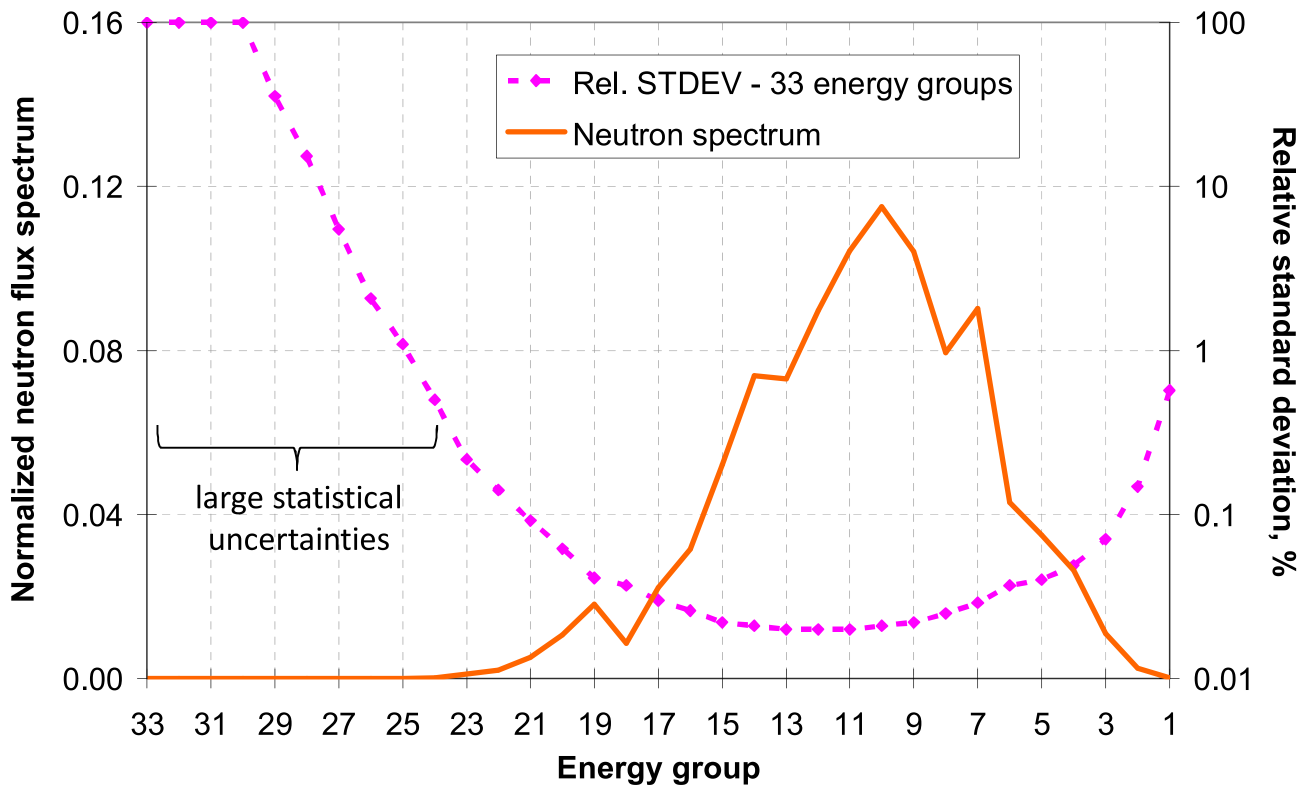
Reference SFR core

- "Working horse" MOX ESFR core design
  - Proposed in the frame of the Collaborative Project on European Sodium Fast Reactor (CP ESFR)



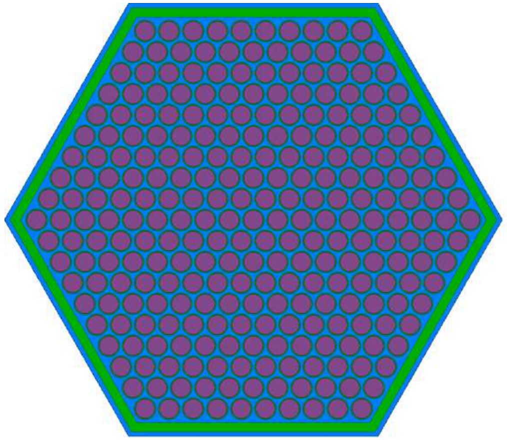
Power density = 206 W/cm^3

Selection of few-group energy structure

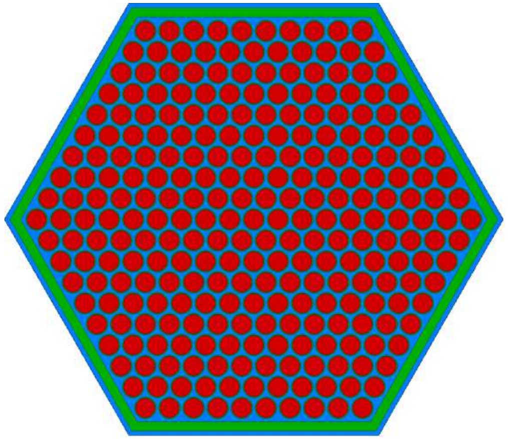


- 33 group structure is not appropriate
  - Very poor statistics in thermal energy groups
- 24 group structure is selected
  - Groups 24 to 33 collapsed into a single thermal group

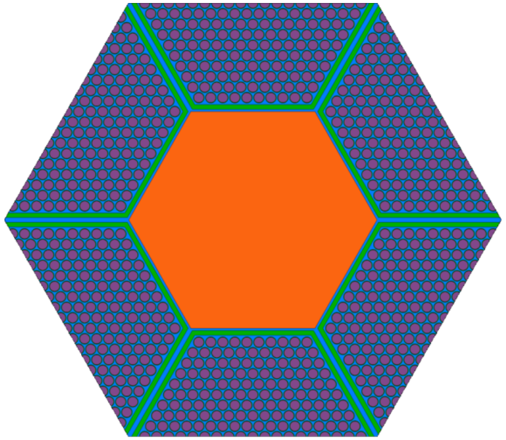
Few-group XS Generation - Super-cell Models



Inner Fuel Sub-Assembly
14.5 wt% Pu MOX



Outer Fuel Sub-Assembly
17.0 wt% Pu MOX

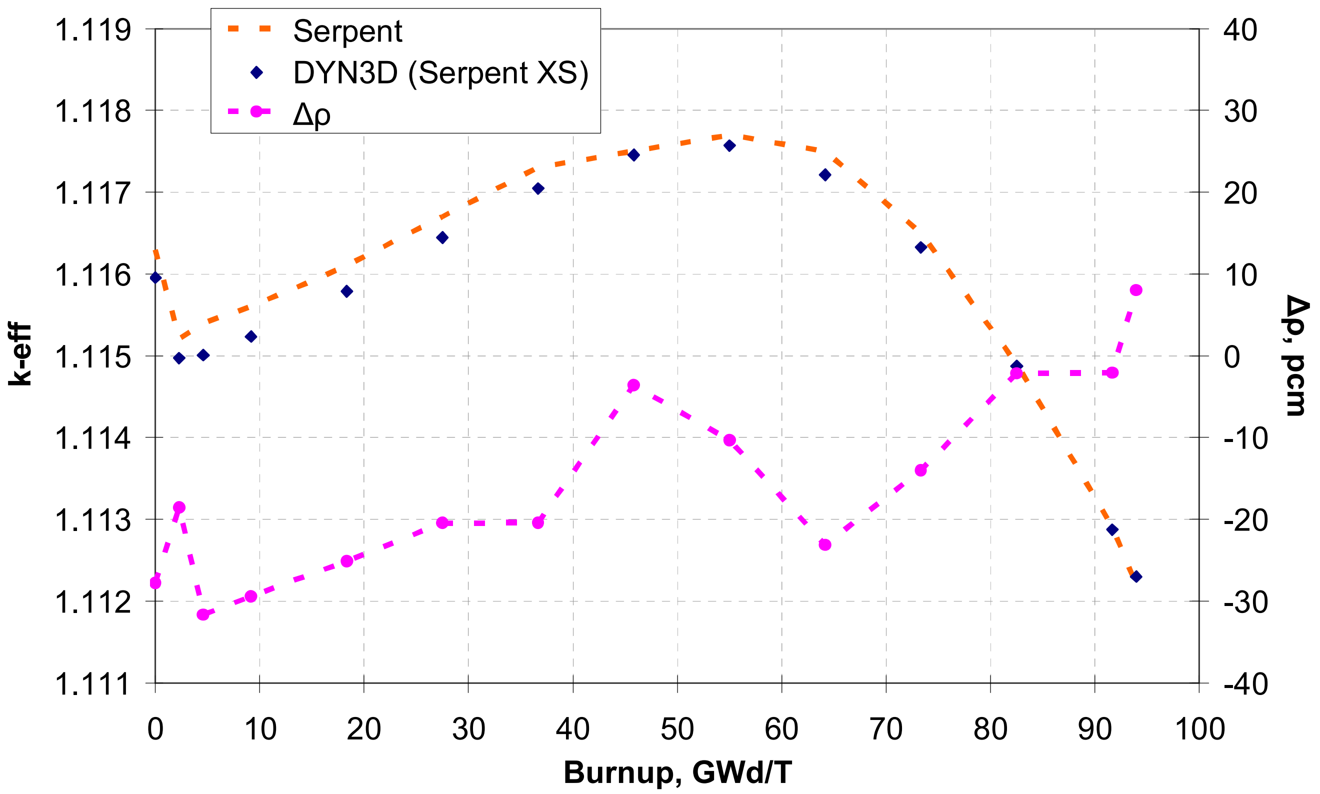


CSD, DSD Model

Radial Reflector Model

Results: k-eff

Full Core



Max. rel. diff. = 32 pcm

Ave. STDEV Serpent = 6 pcm

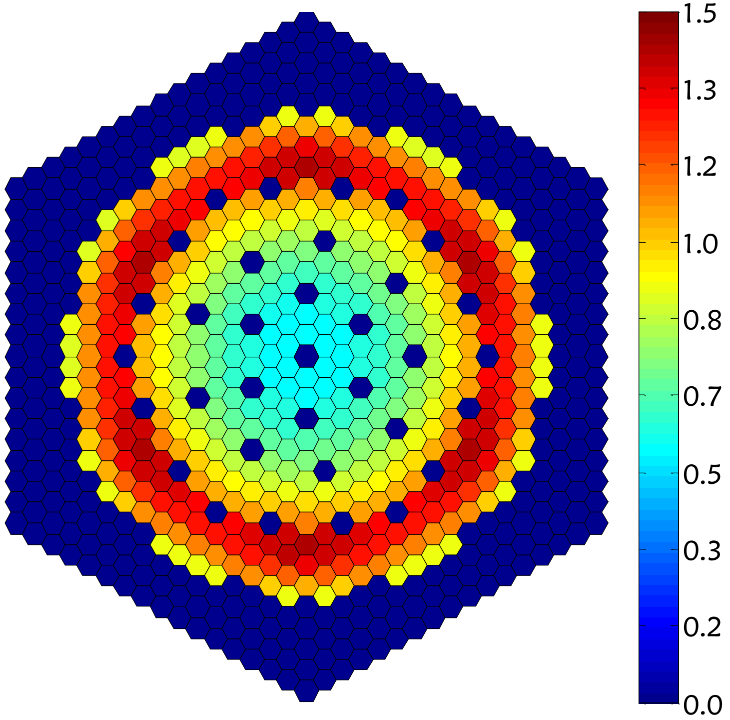
Core integral parameters

Parameter	Stage	Serpent	DYN3D	Diff., pcm Serpent vs. DYN3D
k_D, pcm	BOL	-1062	-1072	**-10**
	EOL	-723	-723	**0**
CVR, pcm	BOL	2821	2850	**29**
	EOL	3654	3702	**47**
Total CDS worth, pcm	BOL	-4678	-4629	**49**

Ave. STDEV k-eff (Serpent) = 6 pcm

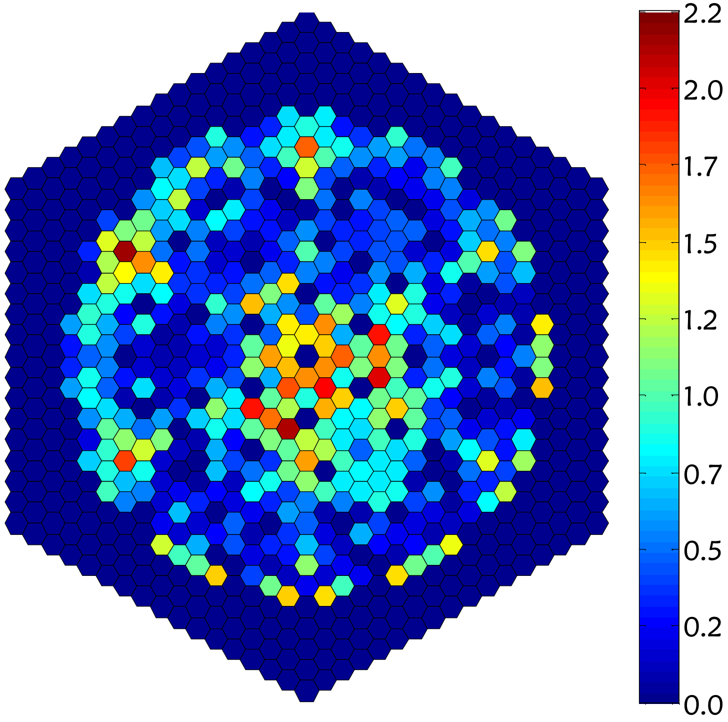
Results: Power Distribution at **BOC**

Control Rods **Out**



Serpent

Ave. STDEV = 1.3 %



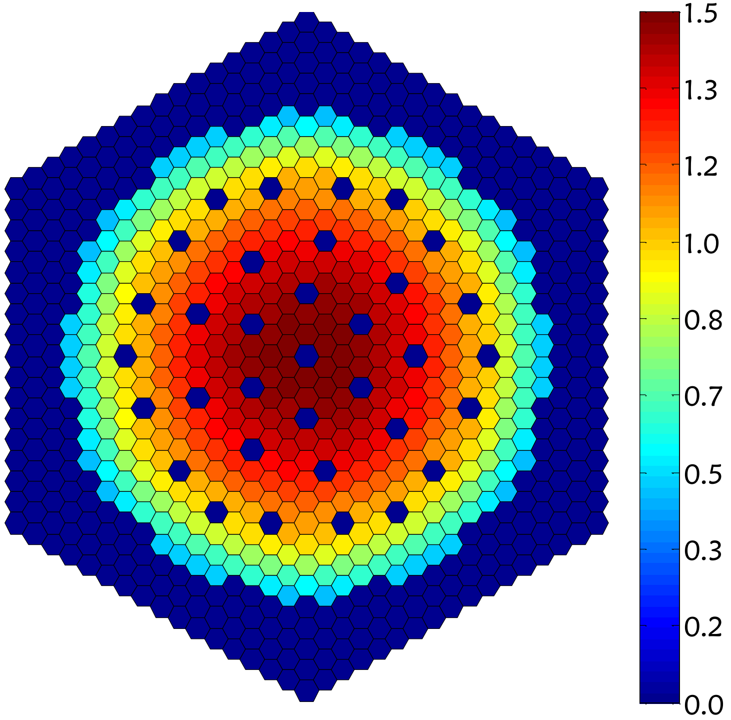
Rel. diff., %
Serpent vs. DYN3D

Max. rel. diff. = 2.1 %

Ave. rel. diff. = 0.6 %

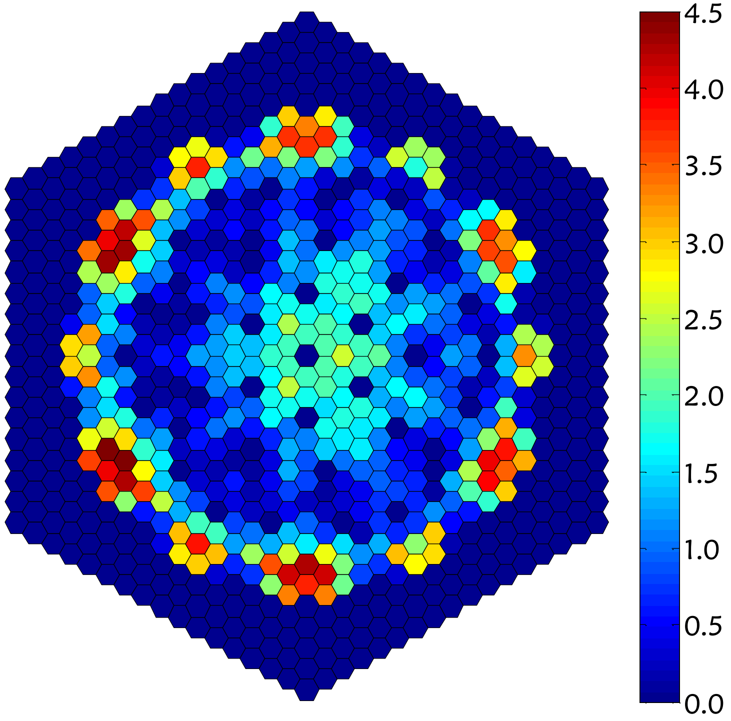
Results: Power Distribution at **EOC**

Control Rods **Out**



Serpent

Ave. STDEV = 1.5 %



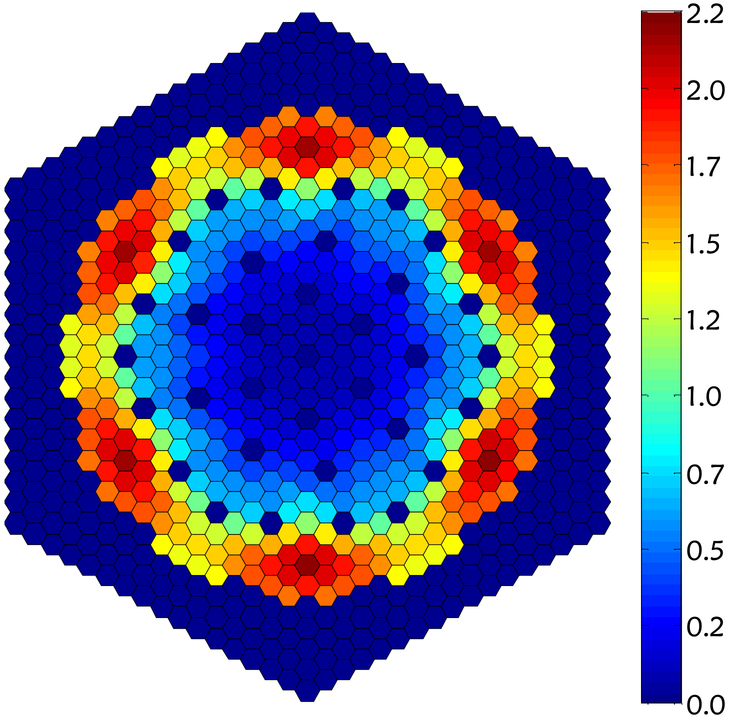
Rel. diff., %
Serpent vs. DYN3D

Max. rel. diff. = 4.5 %

Ave. rel. diff. = 1.4 %

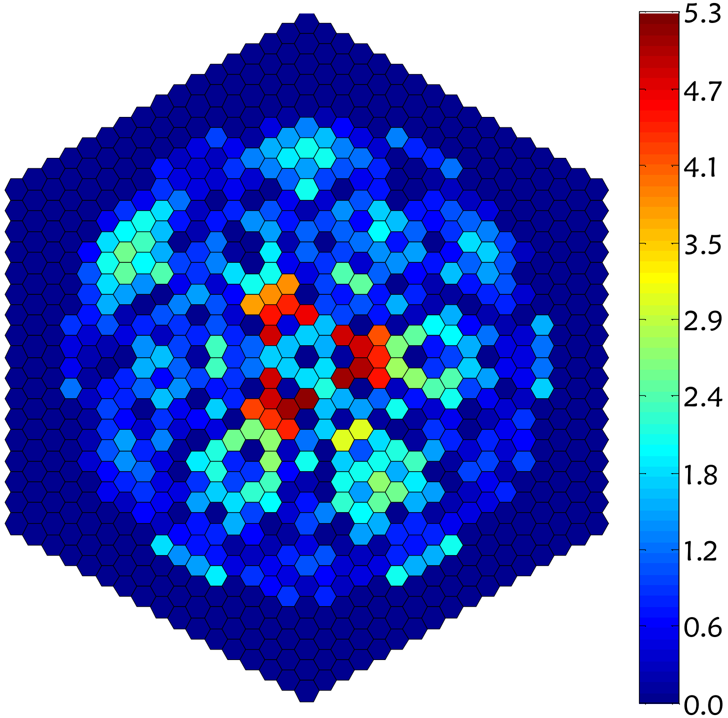
Results: Power Distribution at <u>BOC</u>

Control Rods <u>In</u>



Serpent

Ave. STDEV = 2.1 %



Rel. diff., %
Serpent vs. DYN3D

Max. rel. diff. = 5.2 %

Ave. rel. diff. = 1.2 %

Outlook

- Serpent based few-group XS were used by DYN3D
  - 2D full core nodal diffusion calculations of ESFR core
- DYN3D results were verified against full core Serpent MC solution
- Very good agreement between the codes was obtained
- DYN3D-SFR - required modifications
  - Updating thermal-hydraulic module
  - Development of thermal-mechanical module
- Validation and Verification
  - Benchmarks on EBR and Phoenix experiments

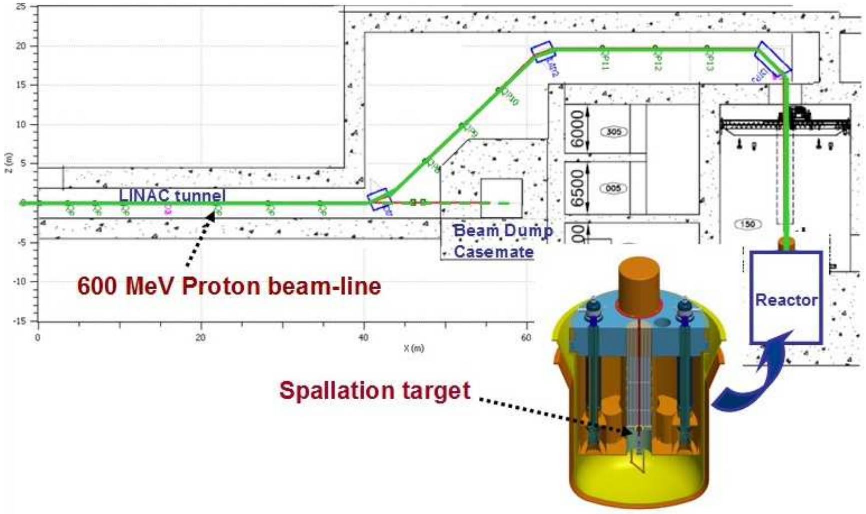
Core shielding assessment for the design of FASTEF-MYRRHA

Anna Ferrari

Work done in the frame of the FP7 European Central Design Team (CDT) Project, which worked to design the FAst Spectrum Transmutation Experimental Facility (FASTEF), to support the construction of MYRRHA

Introduction

- The MYRRHA research facility (SCK·CEN - Mol, Belgium):
  - lead-bismuth eutectic (LBE) cooled reactor
  - working both in critical and in sub-critical operation modes



- One of the many challenges of the MYRRHA design is:
  - shielding of the accelerator tunnel and the reactor building

Shielding and activation analysis

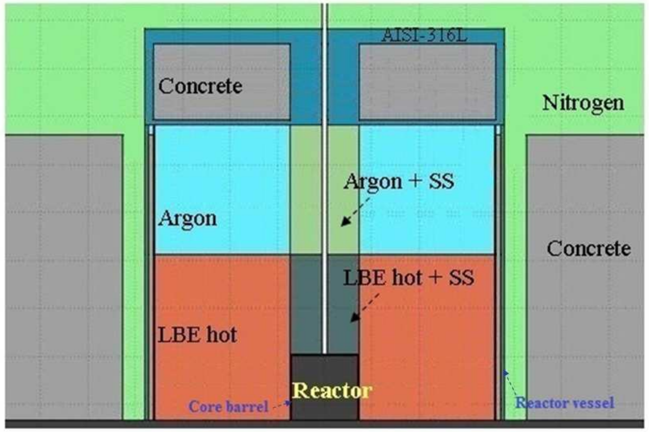
Objectives & Methodology

- **Main goal:**
  - To develop a reliable methodology based on Monte Carlo method for the assessment of the main shielding and activation

- **Analysis methodology:**
  - **MCNPX** CDT models of the MYRRHA critical and subcritical core
    - characterize the neutron radiation fields on suitable surfaces around the core barrel
    - build a complex source terms as an input for the FLUKA simulations
  - **FLUKA** simulations
    - detailed model for shielding and activation analysis
    - using the MCNPX evaluated spectra as a source terms

Shielding and activation analysis

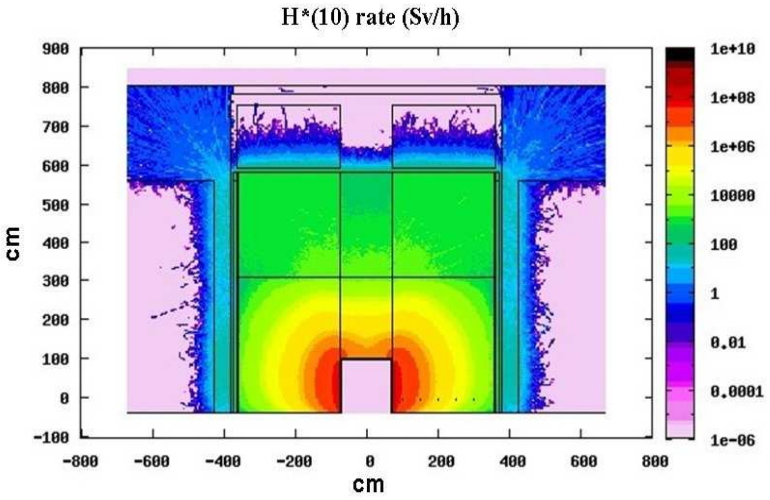
The FLUKA detailed model



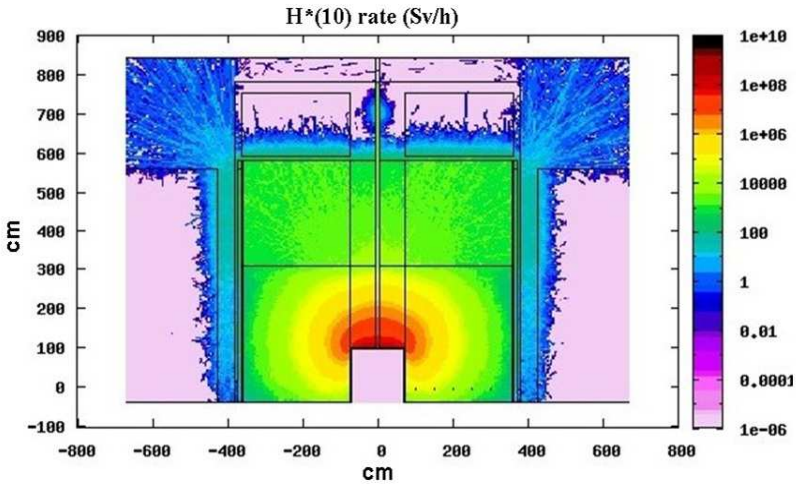
- **Lateral shielding analysis:**
  - FLUKA detailed model from the core barrel to the shielding walls and the reactor cover
  - <u>conservative source term</u>: critical operation mode

- **Vertical shielding analysis:**
  - FLUKA detailed model from the core barrel to the reactor cover and the final wall beyond the last magnet of the proton beam line
  - <u>conservative source term</u>: sub-critical operation mode

Results: Ambient dose equivalent rate



Critical operation mode



Sub-critical operation mode

- The results demonstrated a sufficient lateral radiation containment

Outlook

- A methodology based on combined use of two MC codes (MCNPX and FLUKA) has been developed
    - a powerful tools for shielding optimization of the MYRRHA facility

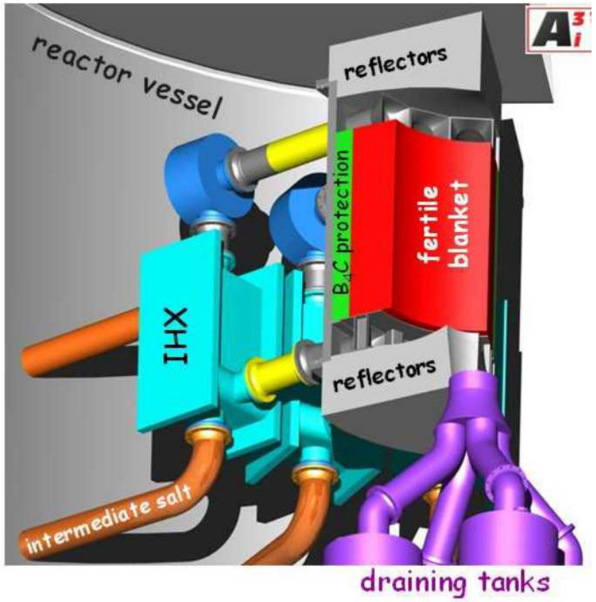
- The developed methodology can address the following key points:
    - optimization of the cover design of the reactor vessel
    - optimization of the upper vertical part of the reactor building
    - choice of structural materials close to the spallation target

Neutron shielding studies on an advanced Molten Salt Fast Reactor (MSFR) design

Bruno Merk, Jörg Konheiser

Introduction

- Advantages of MSFR:
  - online reprocessing and refueling
  - no solid fuel production
  - always negative feedback
  - draining of the fuel

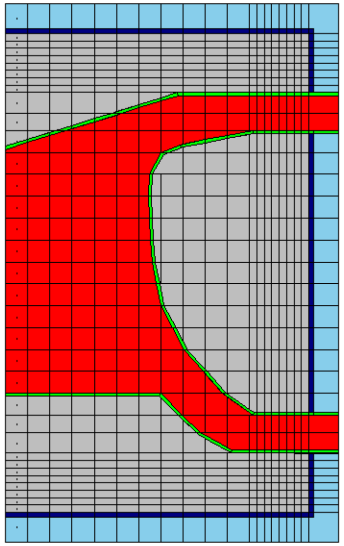


- Material damage in the MSFR is significantly high
  - high neutron flux level in the core
    - high share of neutrons above 1MeV
  - fast neutron are born directly at the core vessel walls

- The neutron fluence in the core and outer vessels should be evaluated

Advanced MSFR model

- Modeling of a 2D advanced MSFR using the HELIOS unstructured mesh neutron transport code

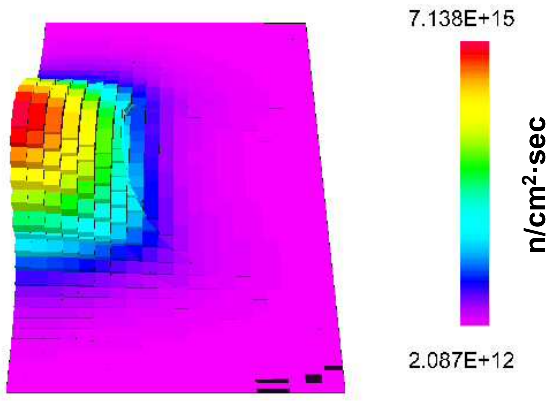


1. **Core region: mixed fuel-fertile salt**
2. **Blanket region: pure fertile salt**
3. **20 mm-thick core vessel**
4. **30 mm-thick outer vessel (safety related vessel)**
5. **20 cm-thick graphite reflector poisoned with 5% natural boron**

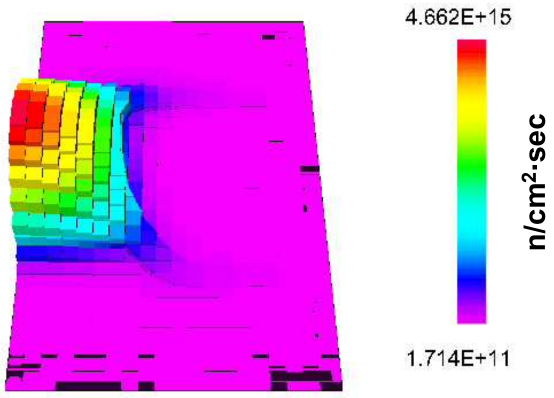
3000 MWth reactor based on the Thorium fuel cycle

Neutron flux distribution

In the core and blanket regions



Epithermal neutron flux (< 0.1 MeV)

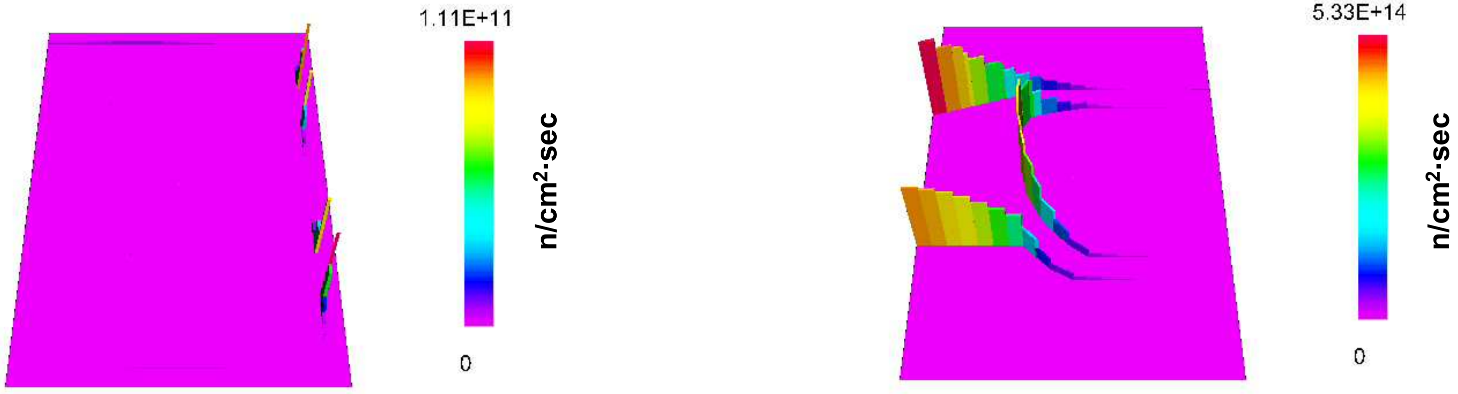


Fast neutron flux (> 0.1 MeV)

- **Fast neutron flux:**
  - strongly bound to the core vessel
- **Epithermal neutron flux:**
  - small change at the boundary between the core and blanket

Neutron flux distribution

In the core vessel walls

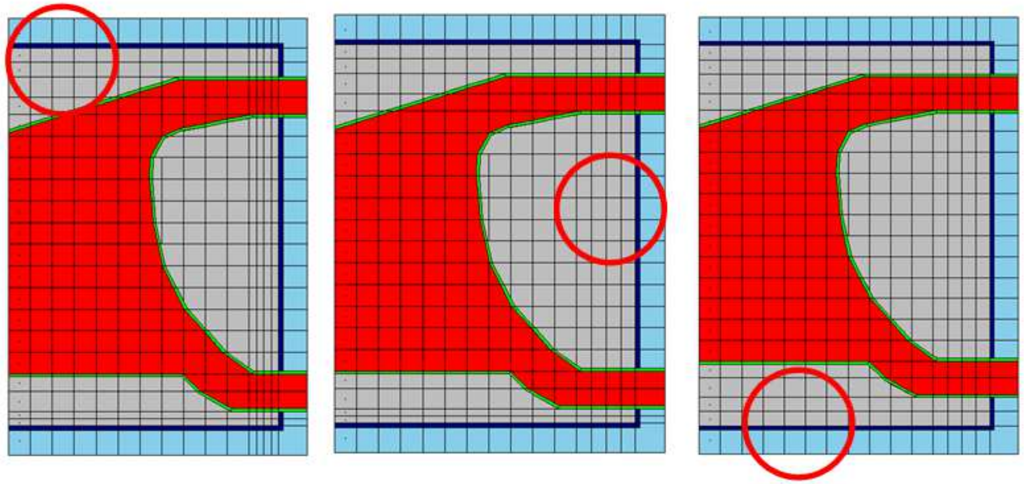


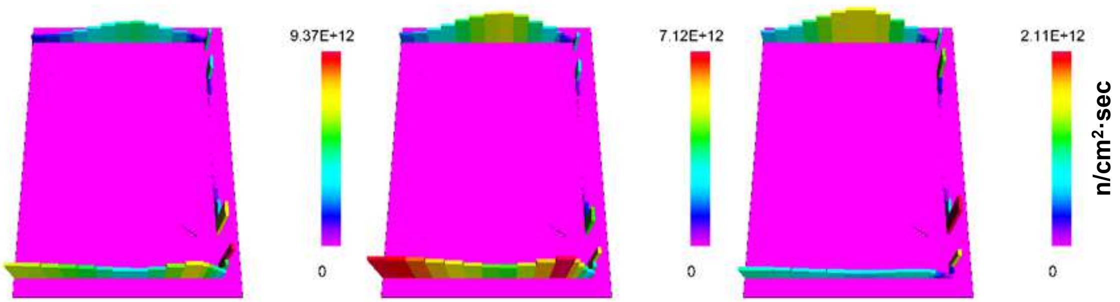
Fast neutron flux (> 1 MeV)

- Very high neutron flux
  - leads to a very short time to reach the fluence limit value for materials under irradiation (10^{20} n/cm^2)
- A failure of the core vessel has to be taken into account

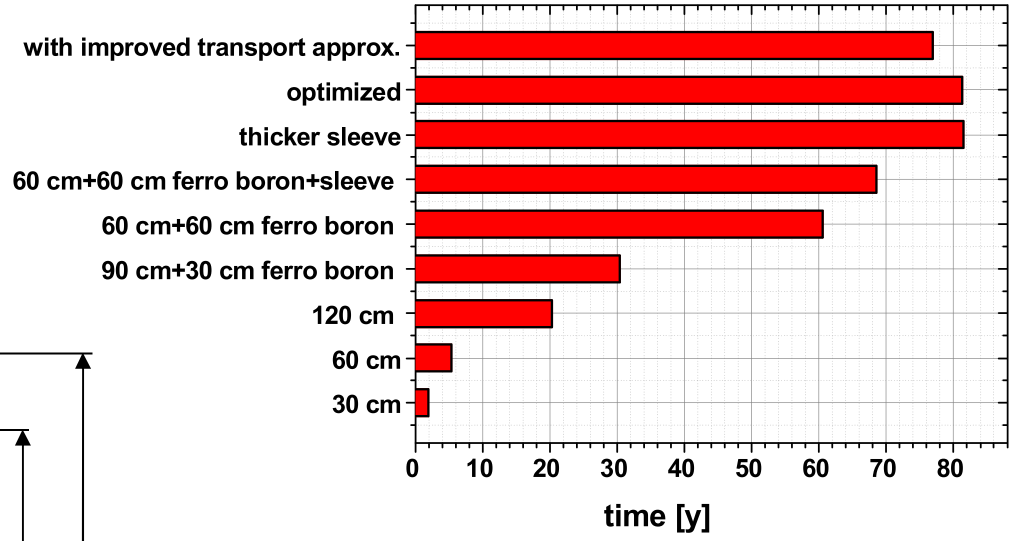
Shielding optimization of the outer vessel

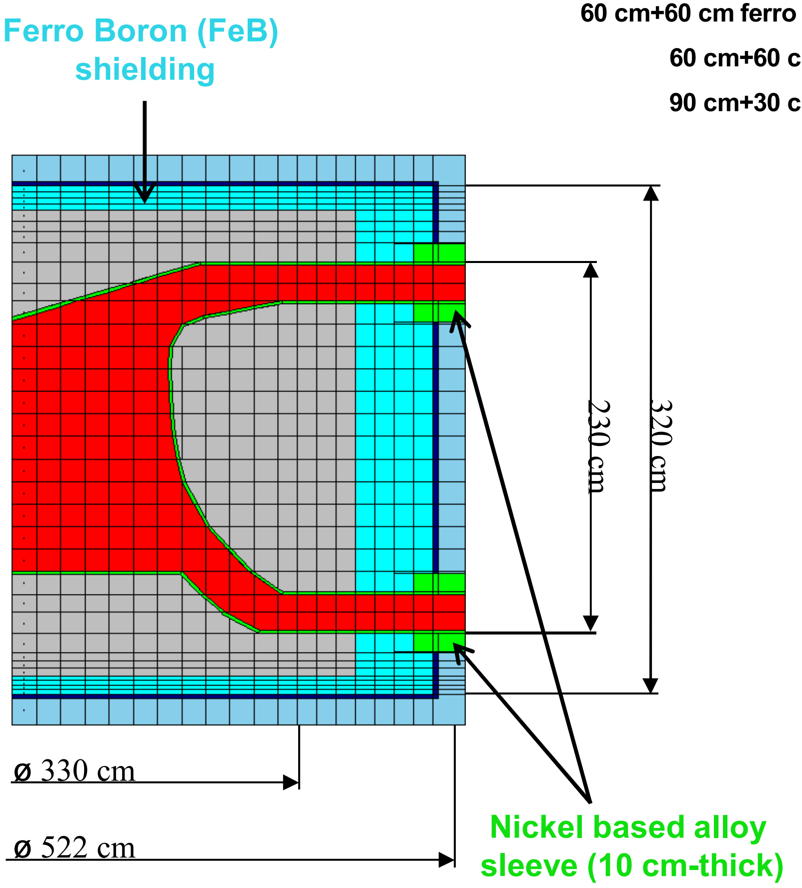
- The outer vessel has to fulfill a safety function
- **Optimization strategy:** stepwise increasing of the blanket region

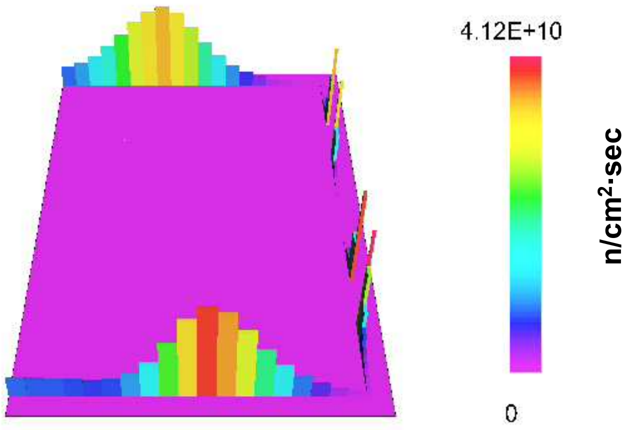




Optimized geometry







Fast neutron flux (> 1 MeV)

Outlook

- A 2D advanced MSFR model was analyzed
  - Using the HELIOS deterministic lattice transport code
- High neutron flux was demonstrated in the core vessel walls
  - The core vessel cannot carry any safety related function
- The outer vessel has to fulfill the safety function
- The outer vessel can be shielded by a well blanketed system
  - keep the fluence within the limit of 10^{20} neutrons/cm^2 for a reasonable operation of 80 years.

Thank you for your attention!