IAEA
International Atomic Energy Agency
Atoms for Peace and Development

Special HTGR topics:
Modelling, Tools and Uncertainty analysis

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Joint IAEA-ICTP Workshop on the Physics and Technology of Innovative High Temperature Nuclear Energy Systems

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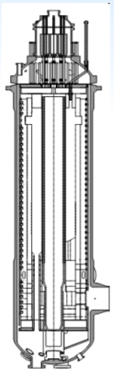
Required Analysis for Design

Engineering Analysis Synergy



NEA to CFD

Reactor Power Profile

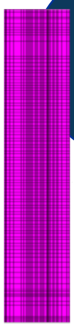
CFD to SAG

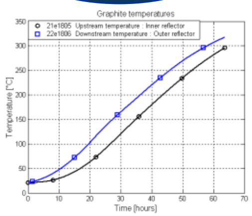
Detailed Component Temperatures

Nuclear Engineering Analysis (NEA)

Computational Fluid Dynamics (CFD)

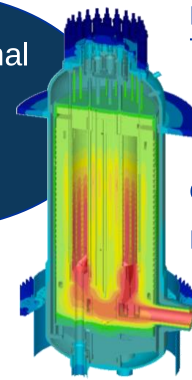
Structural Analysis (SAG)





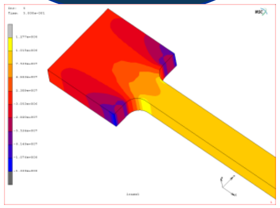
CFD to NEA

Reactor Flow Distribution and Temperatures



CFD to SAG

Fluid/Structure Interaction



CFD to TFAG

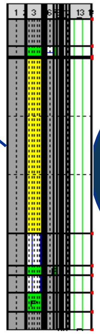
Detailed Flow Distributions

TFAG to CFD

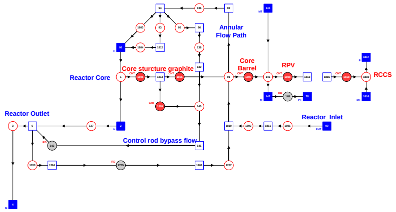
Cycle Flow Conditions

TFAG to/from NEA

Detailed Flow Distributions and Neutronic Data



Thermal Fluid Analysis (TFAG)



Nuclear Engineering Analysis Reactor

Core Neutronics / Thermal Hydraulics

Analyze Neutronic Design for feedback to both engineering and safety

Steady-State (VSOP, MCNP) and Transient (TINTE) Analysis

Input to engineering on core component temperatures, power profiles etc.

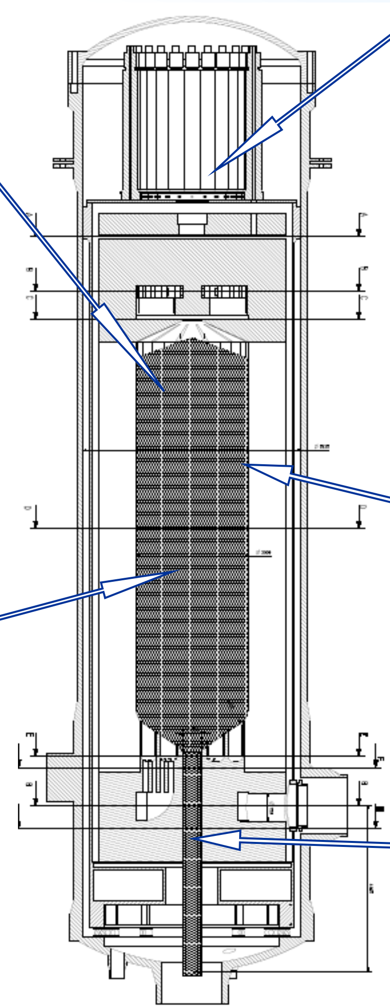
Input to safety on maximum fuel temperatures, control rod worth etc.

Fission Product Releases

Determine Fission Product Releases for both normal operation and accident scenarios

Using Diffusion Theory (GETTER, NOBLEG)

Input to the rest of the source term analysis chain



Shielding and Activation

Analyze Shielding and Activation of core structures and surrounding

Monte Carlo Analysis (MCNP) or simplified transport analysis (MicroShield)

Input to engineering on core component activities for maintenance / decommissioning

Input to safety for worker dose

Dust Generation and Activation

Graphite and metallic dust generation in the core and fuel handling system and activation of the dust

Fuel Source Term

Neutron and Photon source from spent and used fuel

Input to the development of the used and spent fuel tanks and waste handling

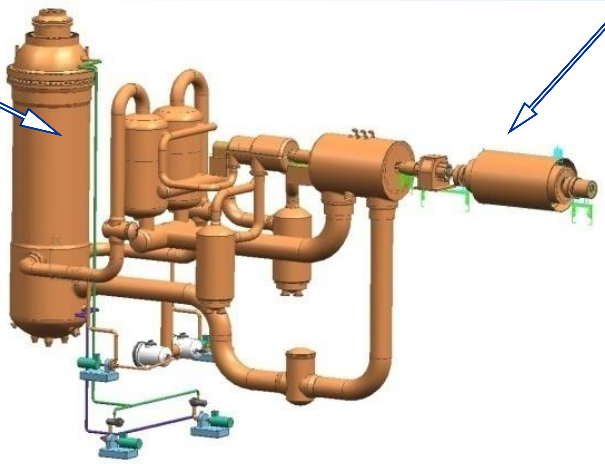
Nuclear Engineering Analysis
Plant

Helium Pressure Boundary Source Term

Analyze the source term within the helium pressure boundary during normal operation and releases in accident scenarios

New code under development (DAMD)

Considers behaviour of dust and fission products in the system



Shielding and Activation

Analyze Shielding and Activation of components

Monte Carlo Analysis (MCNP) or simplified transport analysis (MicroShield)

Input to engineering on component activities for maintenance and decommissioning

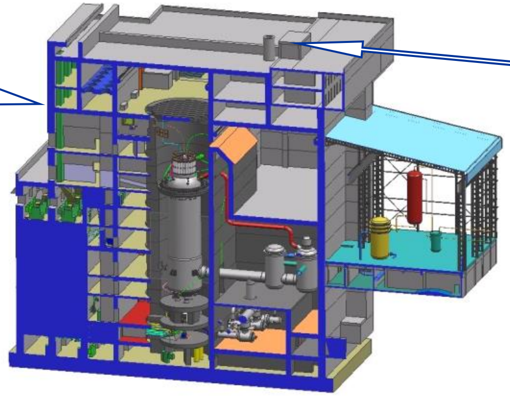
Input to safety for worker dose

Building Retention

Analyze the behaviour of the building for accident scenarios

Accident analysis code (ASTEC)

Input to the building design and safety



Public Dose

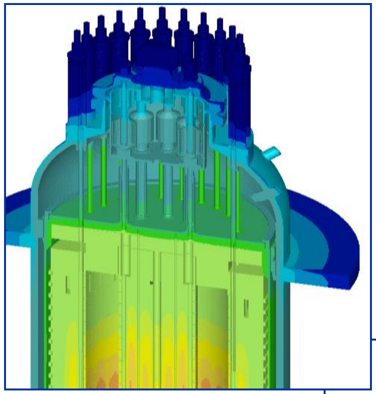
Analyze the expected public dose in accident scenarios

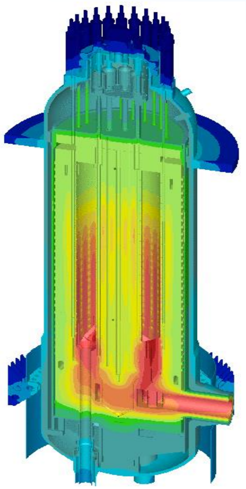
Input to the building design and safety

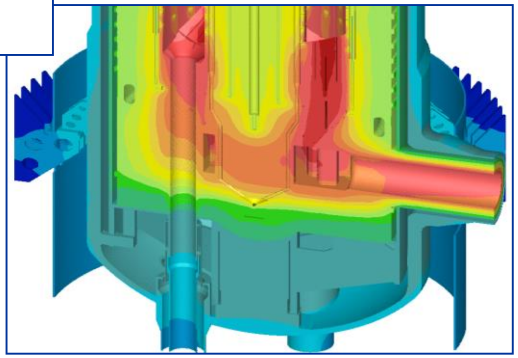
Mechanical Engineering Analysis

Computational Fluid Dynamics

- Responsible for the Detailed Component and Sub-System Analysis of the PBMR Plant
- Expert Use of Commercial and In-House Tools
  - ANSYS Fluent
  - Star-CD/Star CCM+
  - In-House Customisation
- Analysis Functions
  - 3D Simulation of Complex Phenomena
  - Provide Insight to System Behaviour
  - Component Optimisation
  - Input to Safety Analysis



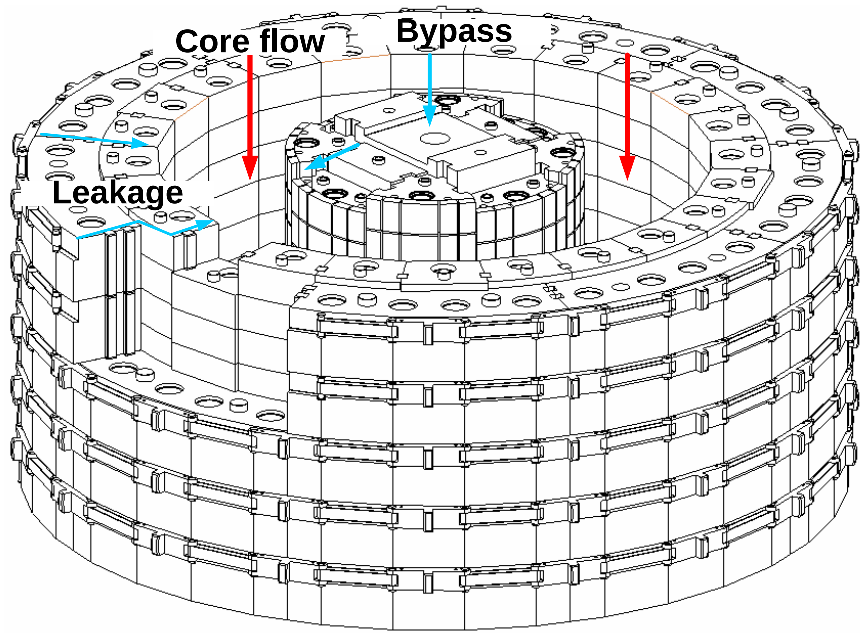


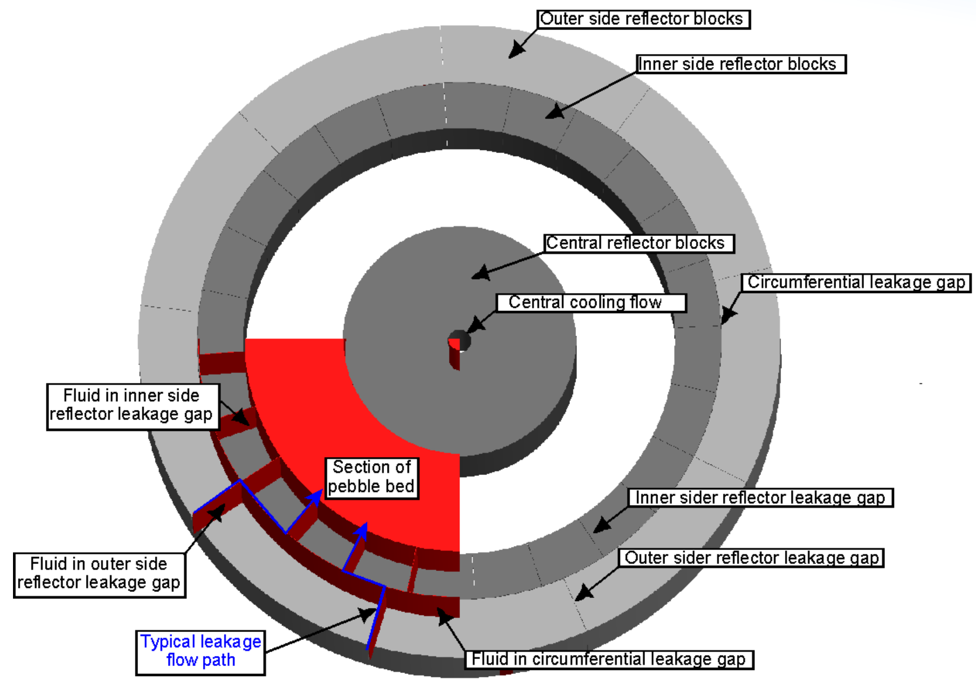


Mechanical Engineering Analysis

Computational Fluid Dynamics

Predicts leak flows used in all reactor codes (Flownex, TINTE, VSOP)

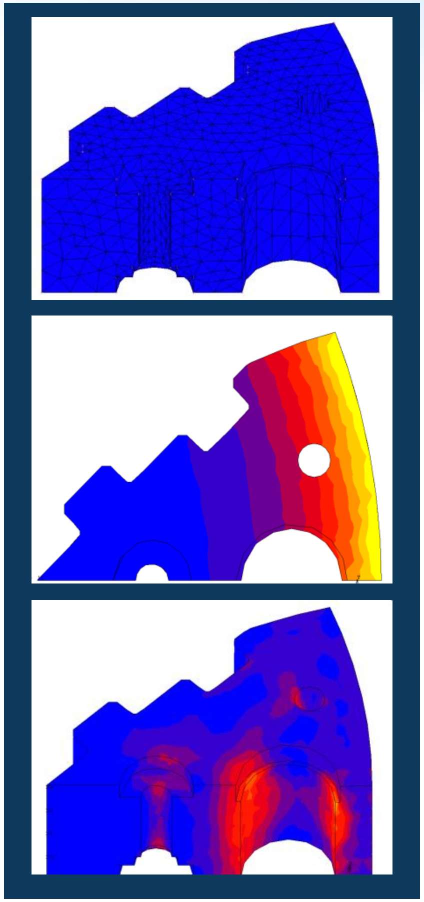




Mechanical Engineering Analysis

Structural Analysis

- o Responsible for the Structural Analysis of the PBMR Plant
- o Expert Use of Commercial Tools
  - ▪ Nastran/Patran
  - ▪ Marc/Mentat
  - ▪ PFC 3D
  - ▪ Dytran
- o Analysis Functions
  - ▪ Modelling Complex Structural Systems
  - ▪ Interpreting Load Information with Respect to ASME Codes
  - ▪ Structural Verification
  - ▪ Input to Safety Analysis



See Paper 158 Geometric layout optimisation of graphite reflector components by Christiaan Erasmus & Michael Hindley

Mechanical Engineering Analysis

Structural Analysis

Sphere Flow Analysis

Nuclear and Mechanical Engineering Analysis Software

Analysis Software Verification and Validation

- V&V plans (and many other supporting documents)
- Procedures accepted by NNR

Software Engineering focus areas:

- Analysis software development and implementation of associated quality and V&V processes
- Legacy analysis software reverse engineering
- Legacy software maintenance
- Methods and software development

Mech Eng Analysis:

ANSYS
Star-CD
Fluent
MSC
Flownex
RELAP
iSight
Matlab
EFD.Lab
PFC3D + 3DEC
Risk Spectrum
HRA Calculator
Fuelnet

Nuclear Eng Analysis:

VSOP
TINTE / MGT
RELAP
NOBLEG
GETTER
RADAX
DAMD
MCNP
ASTEC
SCALE
GENII
AMBER
MicroShield
NJOY
ATILLA
DIREKT
Spectra
MONK
Open Foam
Salome
vulaSHAKA

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Uncertainty Analysis

Introduction

- Traditionally conservative analyses are used for nuclear power plant safety and licensing analyses
- Reliable and high fidelity codes and models allows the use of best-estimate plus uncertainty analysis (as replacement)
- Methodologies to determine the uncertainties in a consistent way have been developed and applied for LWRs
- We need to determine if the same methods can be used for other reactor technologies, where the sources and magnitude of uncertainties are different.
- Also, since very limited experimental results are available for HTGRs some approaches used in the LWR methodology cannot be followed
- The evaluation of uncertainties in high temperature gas-cooled reactors analysis are being investigated within the IAEA coordinated research project

Purpose:

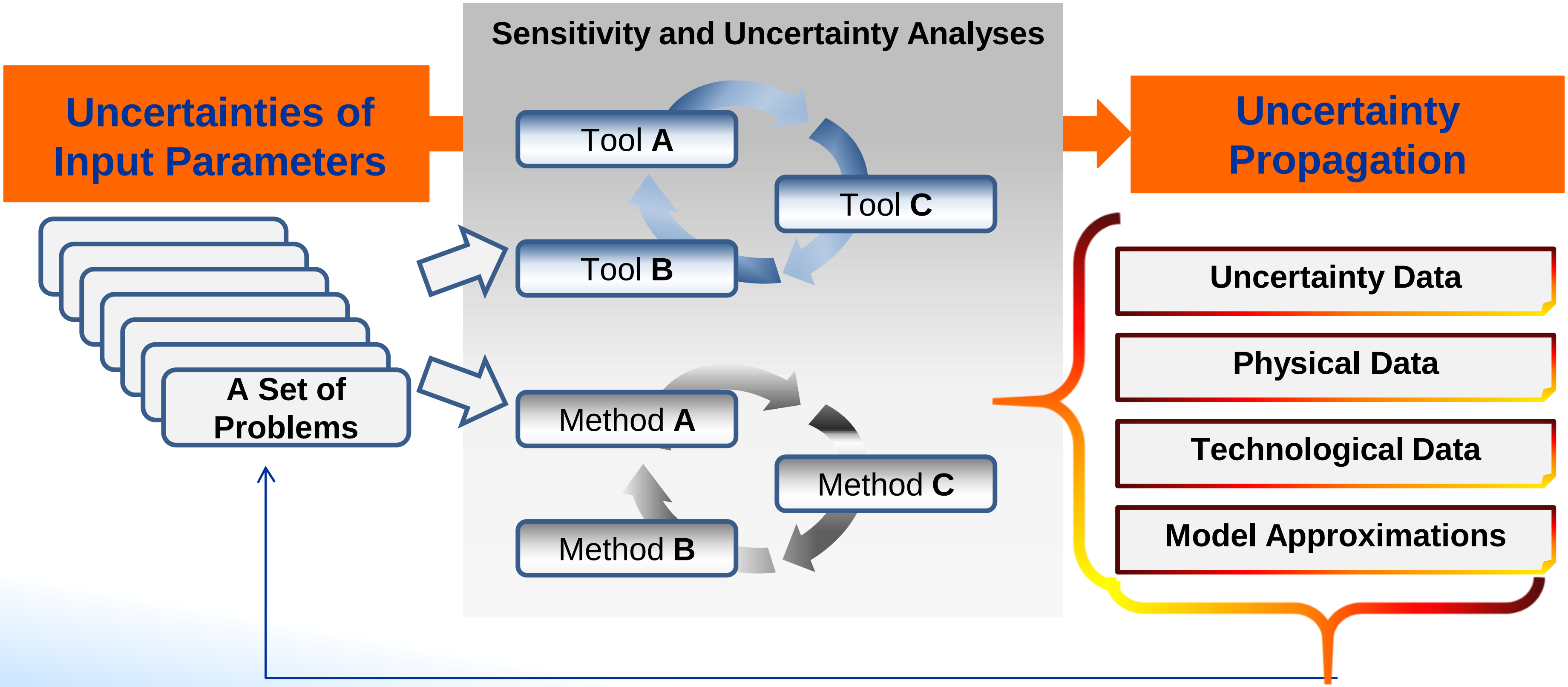
- To determine the uncertainty in HTGR calculations at all stages of coupled reactor physics, thermal-hydraulics and depletion calculations
- Follow the approach of the OECD / NEA UAM LWRs; and aims to:
  - establish and utilize a benchmark for uncertainty analysis in best-estimate coupled HTGR modelling and analysis
  - use as a basis a series of well defined problems with complete sets of input specifications.
  - subdivide the coupled system calculation into several steps, each of which can contribute to the total uncertainty
  - identify input, output, and assumptions for each step.
  - the resulting uncertainty in each step will be calculated (including propagating from previous steps).
  - where possible have reference results and/or experimental results to be used

Why do we expect different results ?

- Compared to LWRs:
  - Different materials
  - Graphite moderator (vs water moderated)
  - Higher operating temperatures ~ Average 900K– 1260K maximum
  - Higher enrichments ~8.5% - 15.5%
  - Higher burnup ~ 100,000 MWd/te
- Modelling aspects
  - Resonance treatment of coated particle fuel (double heterogeneity)
  - Stochastic nature of fuel (particles; pebbles; movement)
  - Much harder neutron spectrum (than PWRs) due to high temperatures and use of graphite moderator
  - long mfp of neutrons exclude the use of simple assembly calculations
    - spectrum calculations need special treatment / in-line / mini-core
    - need multi group core analysis (4, 13, 22 groups)
    - mostly can't use the traditional PWR calculation chain of assembly calculations -> 2-group parameterized library -> core simulator
    - more difficult to propagate uncertainties step-wise – need different approach

CRP on HTGR Uncertainty in Analysis

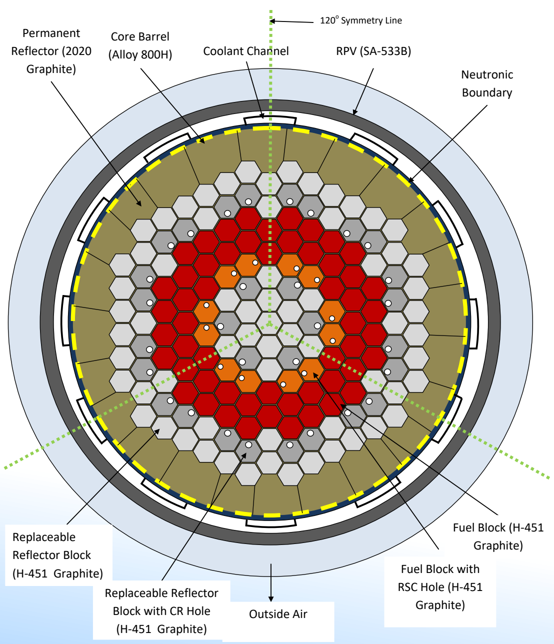
- Objective:
  - To contribute new knowledge towards improving the fidelity of calculation models in the design and safety analysis of high temperature gas-cooled reactors by fully accounting for all sources of uncertainty in calculations.

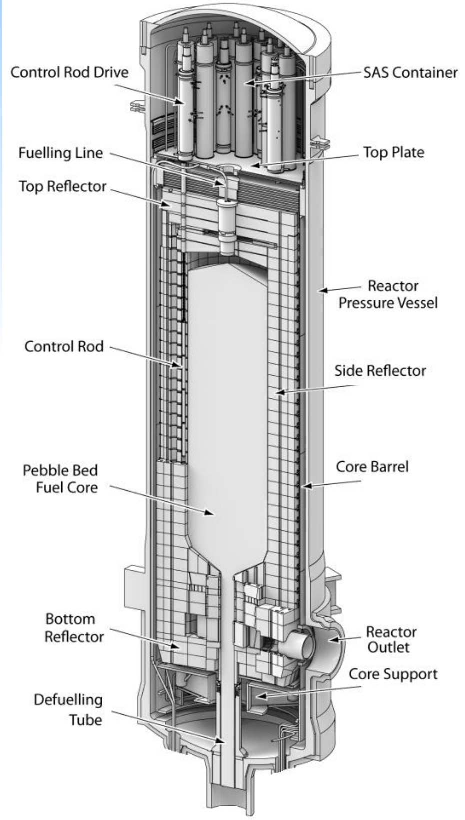


Reference designs

Prismatic Design:

The MHTGR (an earlier General Atomics 350MWth design considered for NGNP) was adopted as the main prismatic reference design.





Pebble Bed Reactor Design:

The HTR-Module-based design, upgraded to 250MWth will be the reference design with some simplifications introduced.

Phases

Phase I
Local stand-alone

- Exercise I-1 & I-2: Local Neutronics; Cell and lattice
- Exercise I-3 & I-4: Local Thermal-hydraulic; SS and transient

Phase II
Global stand-alone

- Exercise II-1:& II-2 Global Core Neutronics (SS and Kinetics)
- Exercise II-3 & II-4: Global Thermal-fluid (SS and transient)

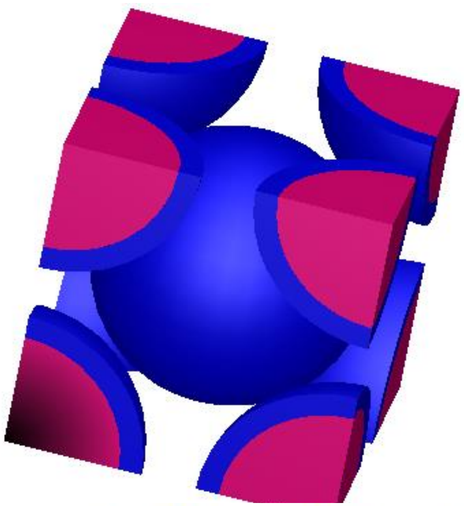
Phase III
Design Calculations

- Exercise III-1: Coupled Steady-state Calculations
- Exercise III-2: Coupled Depletion

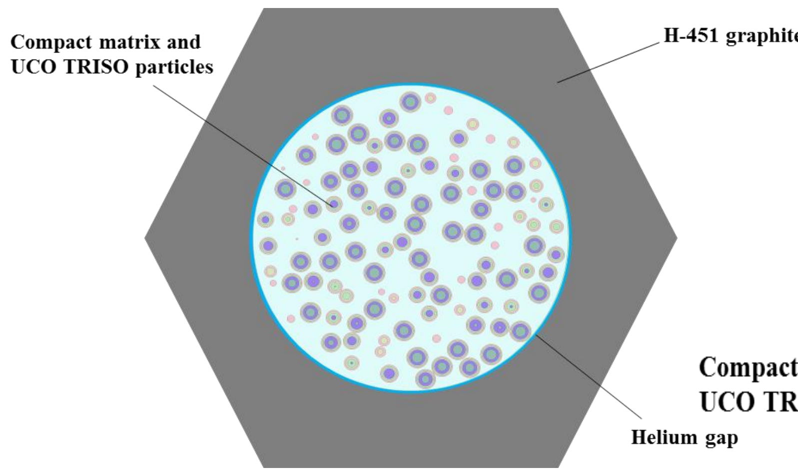
Phase IV
Safety Calculations

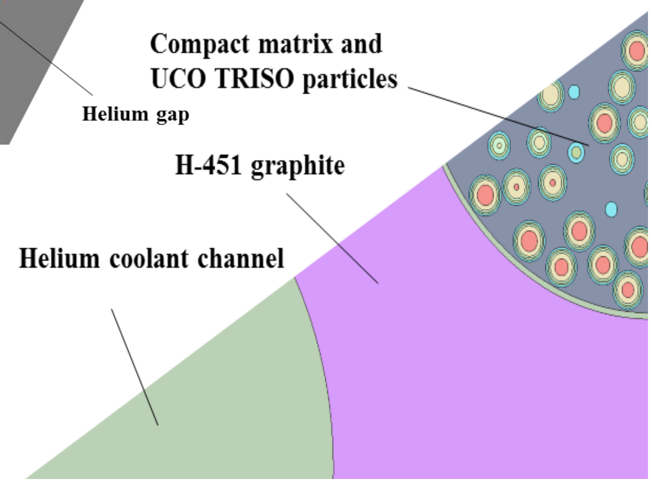
- Exercise IV-1: Coupled Core Transient Calculation
- Exercise IV-2: Coupled System Transient Calculation

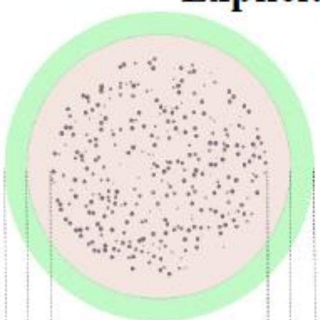
Selection of Preliminary Results:
Phase I: Simplified cell calculations



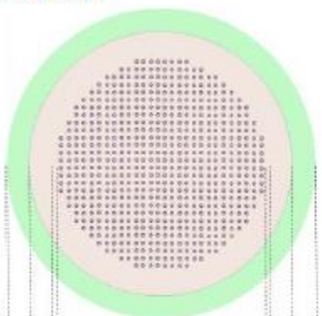
Explicit particle model







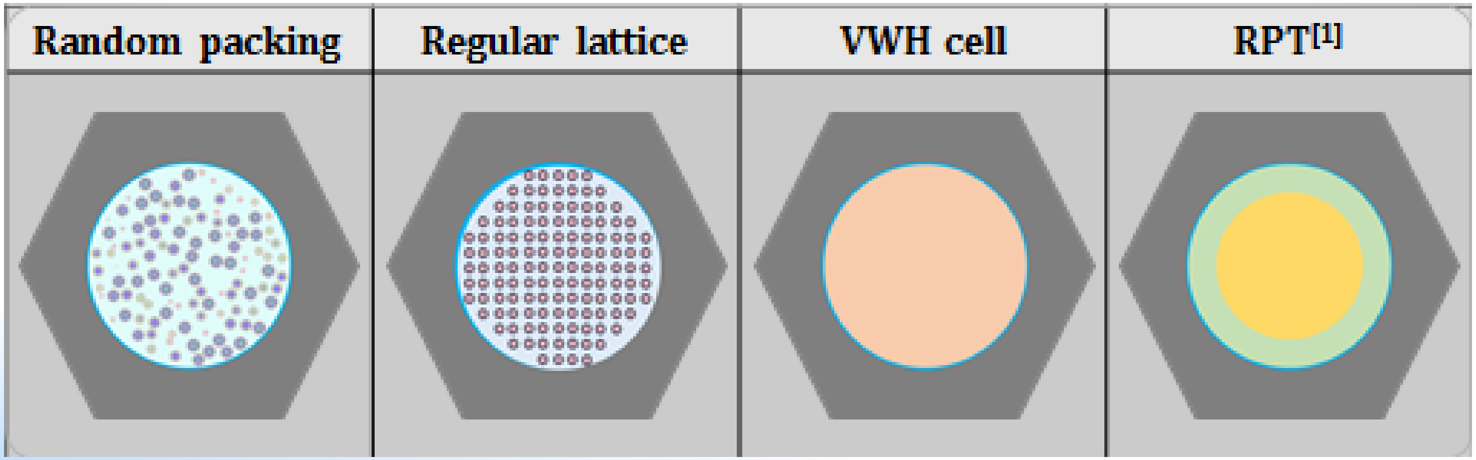
Random model



Lattice model

Importance of correctly modelling the double heterogeneity

- Results show that the correct resonance treatment is required to model the double heterogeneity (I: small kernels and II: fuel compacts in blocks / pebbles)
  - Need explicit random models or regular lattice definition of the fuel kernels, or
  - Approximate models with the Reactivity-Equivalent Physical Transformation (RPT) or use the two-step DOUBLE_HET approach as implemented in SCALE



Model effects

- Large difference in reactivity (always known):

SERPENT HFP	Difference = Random - #, # = Lattice, Homogenized, RPT, unit [pcm]			
	Random	Lattice	VWH	RPT
Hexagonal	1.25239 ± 0.00013	610± 13	5446 ± 20	45 ± 19
Triangular	1.31492 ± 0.00013	-641 ± 17	6325 ± 20	59 ± 18

- Noticeable differences in uncertainty estimate

	Relative standard deviation of k_{inf} (%Δk/k) due to cross section covariance data	
	CZP	**HFP**
Ex. I-1a (VWH)	0.527 ± 0.0002	0.579 ± 0.0003
Ex. I-1b (RPT)	0.508 ± 0.0002	0.547 ± 0.0003

- RPT gives similar results than reference

Further development in methods, updated cross section libraries and covariance matrices

Example:

- SCALE6.2. has been released in April 2016
- ENDF-B/VII.0 and ENDF-B/VII.1 are available in SCALE6.2
- Comparison for KENO-VI
  - Updated cross section library
  - Updated cross section covariance data

Further development in methods, updated cross section libraries and covariance matrices

- CZP – Fresh Pebble cell

	SCALE6.1		SCALE6.2		Code update	SCALE6.2		XS Update
	ENDF7.0[a*]		ENDF7.0[b]		Diff. (b-a)	ENDF7.1[b*]		Diff. (7.1-7.0)
MG Lattice	1.58546	14	1.58997	14	$\boxed{451}$	1.57097	13	-1449
CE Lattice	1.58628	14	1.58653	12	25	1.57932	14	-696
MG DH	1.57744	12	1.58333	12	$\boxed{589}$	1.57636	11	-108
CE RPT	1.58533	13	1.58518	18	-15	1.57867	15	-666
MG RPT	1.58517	14	1.58598	14	81	1.57869	13	-648

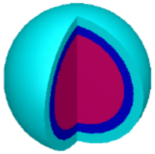
*a: Calculated by KENO-VI in SCALE6.1, *b: Calculated by KENO-VI in SCALE6.2

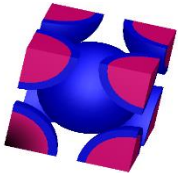
- There was an improvement of MG resonance self-shielding treatment methodology in SCALE6.2.
- Reactivity reduction (~600 pcm) is observed in ENDF/B-VII.1 because of change in graphite XS

Multiplication Factor Uncertainty Results

(With Double heterogeneity modelled as explicit regular lattice)

Case/Model	TSUNAMI/KENO-VI lattice	
	SCALE6.1 ENDF/B-VII.0 % Δk/k	SCALE6.2 ENDF/B-VII.1 % Δk/k
Exercise I-1a Fresh CZP	0.46	0.50
Exercise I-1a Fresh HFP	0.47	0.51
Exercise I-2b (+ high burnup) CZP	0.59	0.53
Exercise I-2b (+ high burnup) HFP	0.67	0.53
Exercise I-2c (+ Fresh) CZP	0.45	0.48
Exercise I-2c (+ Fresh) HFP	0.47	0.47
Exercise I-2d (+ reflector) CZP	0.55	0.54
Exercise I-2d (+ reflector) HFP	0.60	0.50





OLD: Uncertainties increase with temperature, burnup

Library effects and changes due to SCALE6.2 evaluated

Manufacturing uncertainties

IAEA

- Manufacturing uncertainties derived from the fuel used in the ASTRA facility
- Can be modelled in SAMPLER / SCALE6.2 in combination with the cross section covariance, or separately
- Variations may be correlated or bounded by other fuel parameters, i.e. total number of kernels in a pebble will be bounded by mass of U loaded

CRP: Example of manufacturing variation

Description	1σ uncertainty (%)
Fuel pebble	
Outer radius of fuel pebble	± 0.03%
Radius of inner fuel zone	± 0.60%
Packing fraction of fuel pebble	± 0.26%
Heavy metal mass in pebble	± 0.082%
Density of graphite matrix in pebble core and pebble shell	± 0.16%
Density of graphite reflector	± 1.18%
RPT radius only for RPT model	RPT radius table
TRISO particle	
Fuel kernel radius	± 0.98%
Porous carbon layer thickness	± 7.45%
IPyC layer thickness	± 5.56%
SiC layer thickness	± 1.96%
OPyC layer thickness	± 1.75%
UO_2 fuel enrichment	± 0.14%
UO_2 kernel density	± 0.10%
Porous carbon layer density	± 2.97%
IPyC layer density	± 1.54%
SiC layer density	± 0.92%
OPyC layer density	± 1.59%

Single fuel parameter perturbation tests

Parameter	-1sigma	-1/2sigma	Unperturbed	+1/2sigma	+sigma	(Max-Min)
Kernel radius	1.57533	1.57588	1.57601	1.57655	1.57752	219
1st Layer thickness	1.57659	1.57653	1.57625	1.57617	1.57594	65
2nd Layer thickness	1.57635	1.57618	1.57625	1.57622	1.57618	17
3rd Layer thickness	1.57643	1.57631	1.57640	1.57636	1.57602	41
4th Layer thickness	1.57630	1.57617	1.57625	1.57630	1.57639	22
Pebble-core radius	1.57683	1.57643	1.57625	1.57577	1.57577	106
Pebble radius	1.57605	1.57632	1.57625	1.57613	1.57645	40
Heavy metal mass	1.57671	1.57646	1.57601	1.57636	1.57618	70
Kernel mat. fraction	1.57613	1.57629	1.57601	1.57653	1.57646	52
1st Layer	1.57621	1.57642	1.57625	1.57655	1.57642	34
2nd Layer	1.57646	1.57617	1.57625	1.57615	1.57629	31
3rd Layer	1.57627	1.57639	1.57625	1.57615	1.57632	24
4th Layer	1.57629	1.57627	1.57625	1.57636	1.57641	16
Graphite	1.57575	1.57603	1.57625	1.57644	1.57641	69
Pebble PF	1.57627	1.57636	1.57625	1.57623	1.57645	22

Input variations taken from the ASTRA specification

Manufacturing uncertainty results

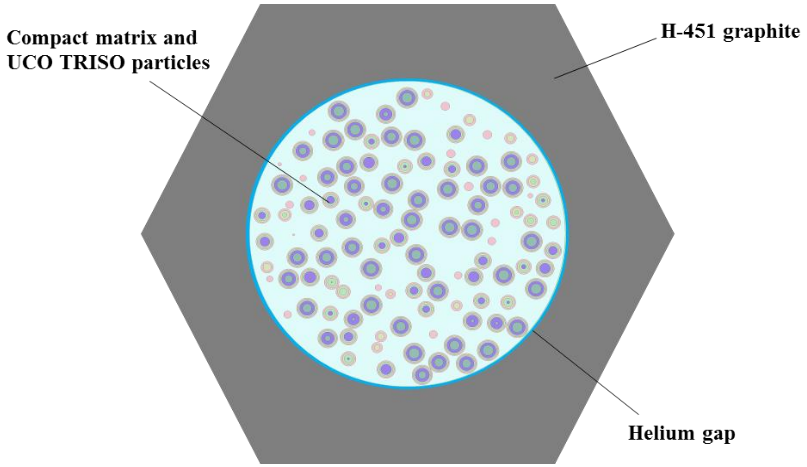
- Initial 130 runs (Wilk's formula)
  - Need to perform more calculations to confirm / input uncertainties / parameters may not be independent

Description			Avg. keff	std. of keff of samples
Ex.I-1a	DOUBLEHET	CZP	1.5763	133 pcm (0.08%)
		HFP	1.4991	150 pcm (0.10%)
Ex.I-1b	DOUBLEHET	CZP	1.1054	111 pcm (0.10%)
		HFP	1.0706	120 pcm (0.11%)

- First results obtained seems to be consistent with the parameter study
- The uncertainties introduced are substantially smaller than the contribution of the cross section uncertainties

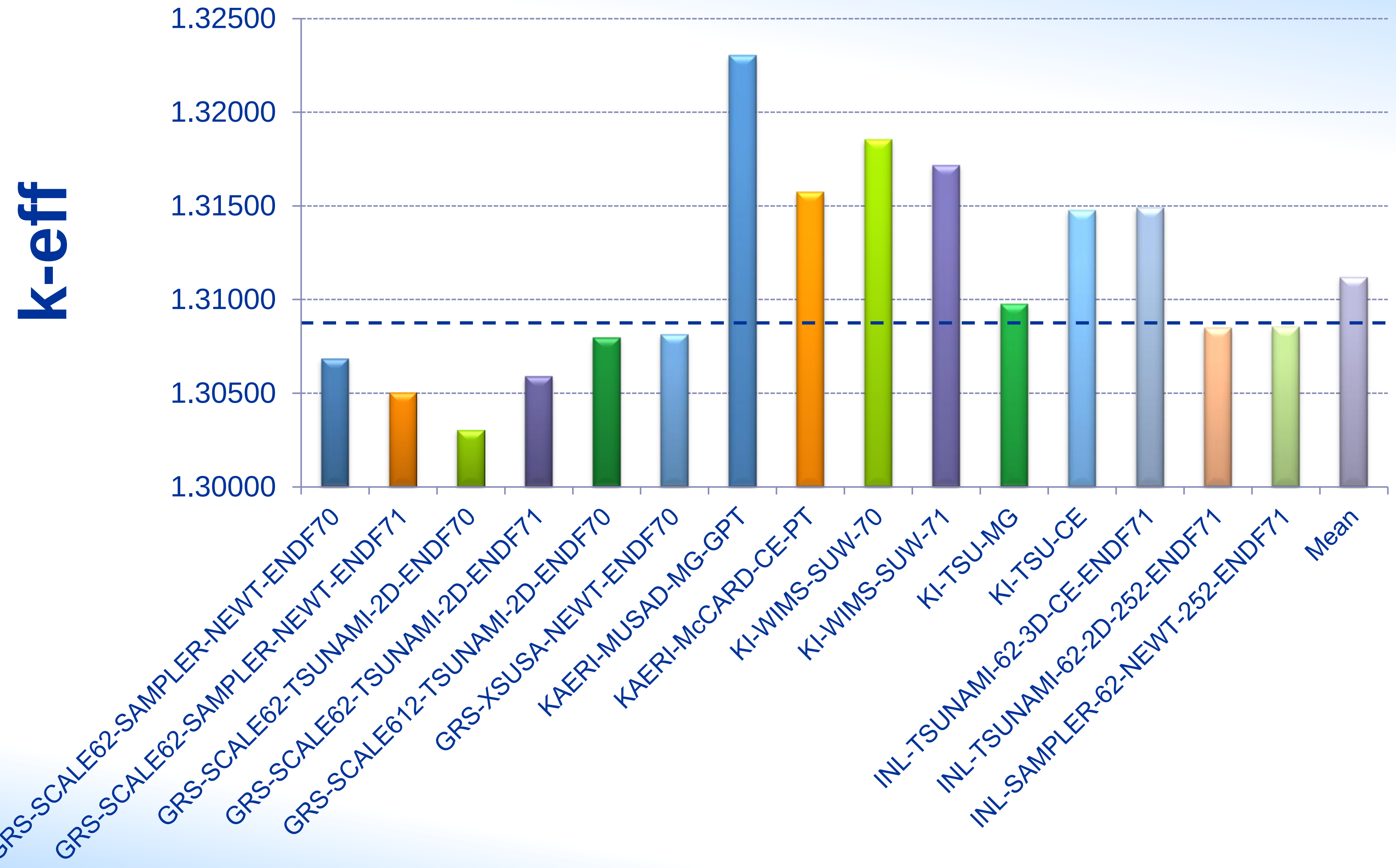
Comparative results: Prismatic cell calculations

- Prismatic compact with surrounding graphite



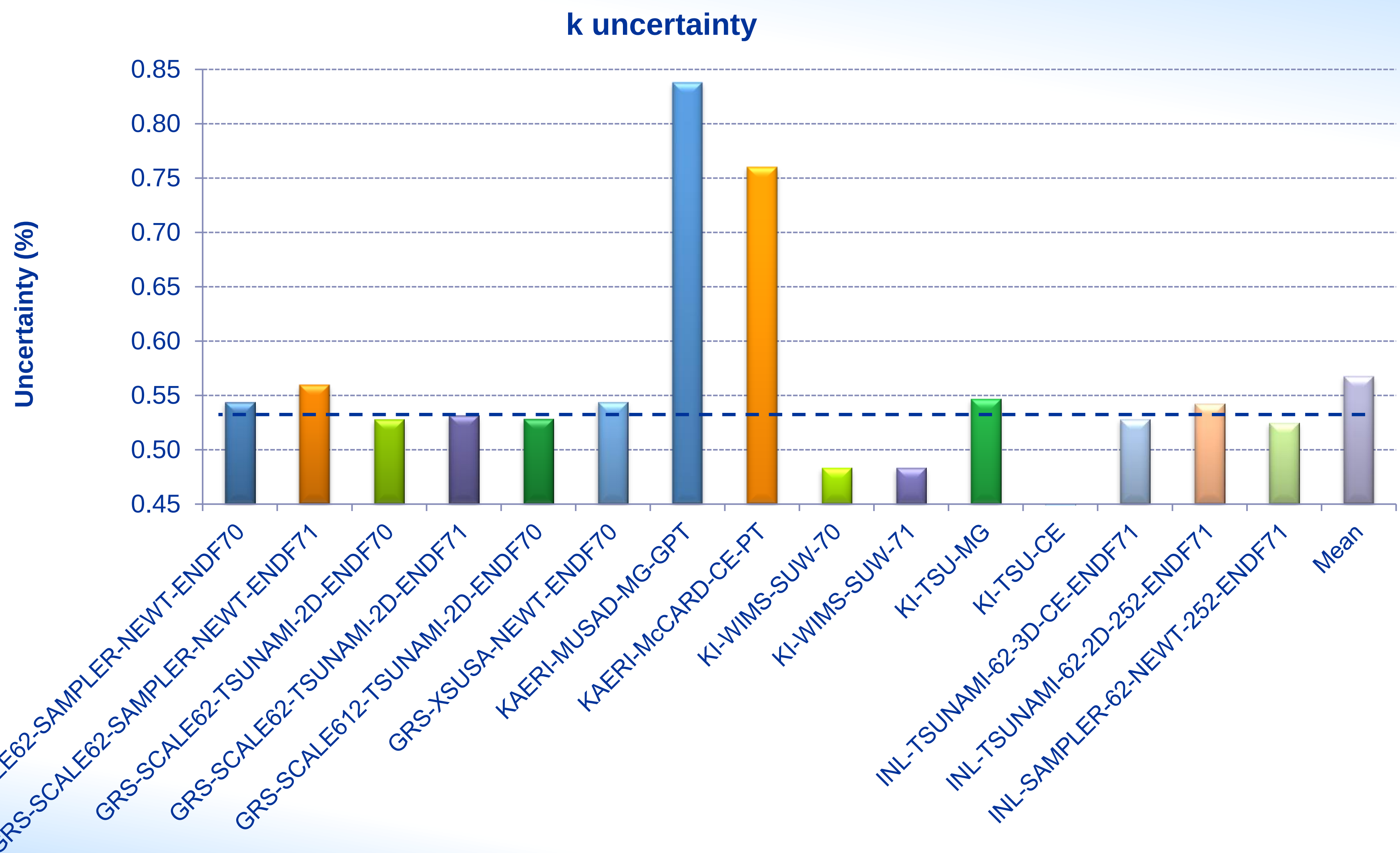
- First (unverified) results submitted (blind calculations)
  - Eigenvalue
  - Uncertainties
  - Contributors to uncertainty
  - 1-group cross section

Eigenvalue

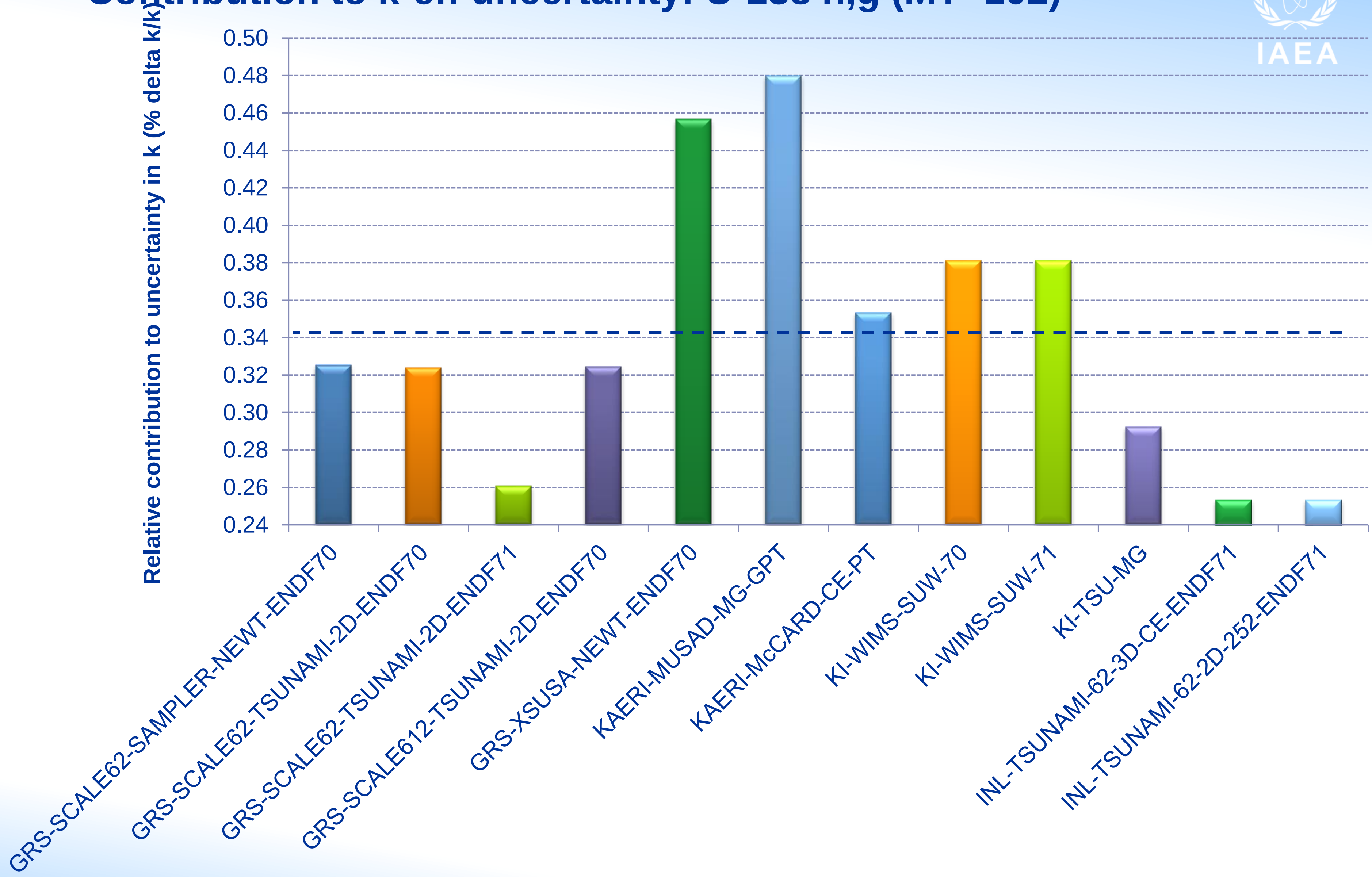


k-eff
1.32500
1.32000
1.31500
1.31000
1.30500
1.30000
GRS-SCALE62-SAMPLER-NEWT-ENDF70
GRS-SCALE62-SAMPLER-NEWT-ENDF71
GRS-SCALE62-TSUNAMI-2D-ENDF70
GRS-SCALE62-TSUNAMI-2D-ENDF71
GRS-SCALE612-TSUNAMI-2D-ENDF70
GRS-XSUSA-NEWT-ENDF70
KAERI-MUSAD-MG-GPT
KAERI-McCARD-CE-PT
KI-WIMS-SUW-70
KI-WIMS-SUW-71
KI-TSU-MG
KI-TSU-CE
INL-TSUNAMI-62-3D-CE-ENDF71
INL-TSUNAMI-62-2D-252-ENDF71
INL-SAMPLER-62-NEWT-252-ENDF71
Mean

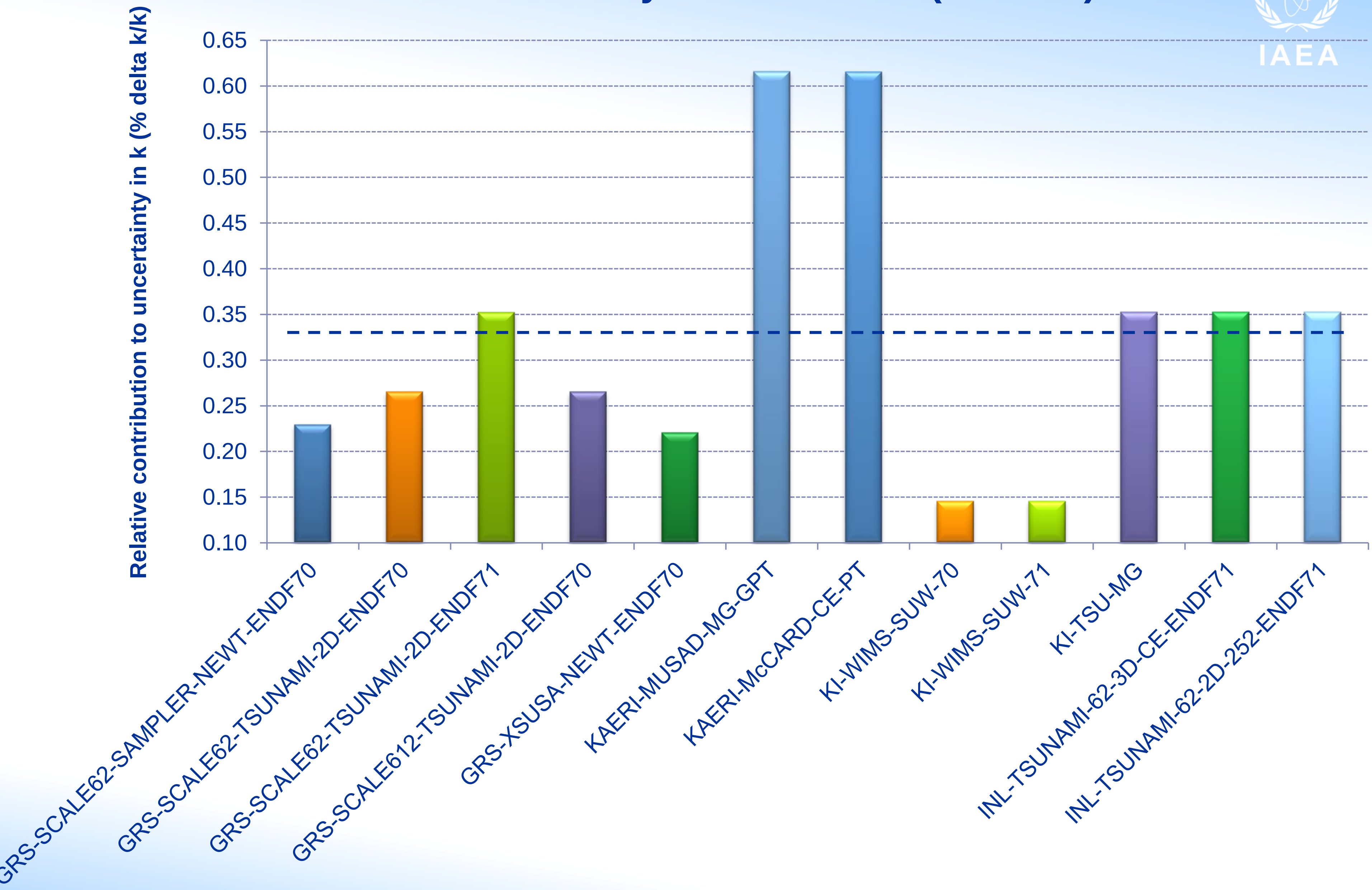
Eigenvalue uncertainty



Contribution to k-eff uncertainty: U-238 n,g (MT=102)



Contribution to k-eff uncertainty: U-235 nu-bar (MT=452)



Summary for Phase I

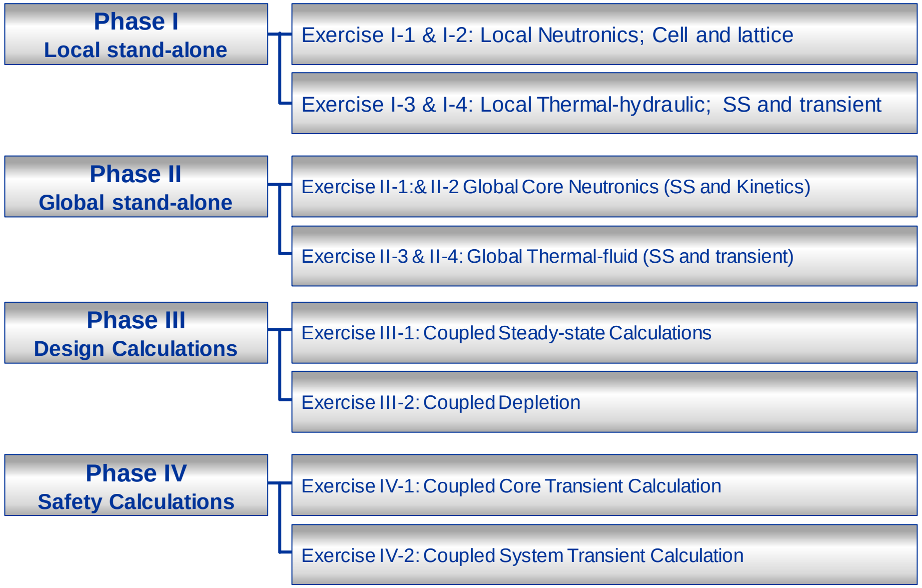
- Uncertainties in calculated k-eff (due to cross section uncertainties)
  - Similar but slightly larger than for LWR / thermal systems
  - All effects due to models, libraries and covariance sets to be quantified
- Proper treatment of the double heterogeneity is required to correctly determine the contribution of cross section uncertainties to k-eff
  - Uncertainties calculated with the Reactivity-Equivalent Physical Transformation method show good agreement
- Some of the top five contributors identified also found to contribute to the uncertainties in light water reactor test cases

  $^{238}U(n,\gamma)$, $^{235}U_{(nubar)}$, $^{235}U(n,\gamma)$, $^{235}U_{(fission)}$, $^{239}Pu_{(nubar)}$

  but others, $^{135}Xe(n,\gamma)$ and graphite capture or elastic scattering.
- Comparing results and updated SCALE 6.2
  - New covariance matrices / Updated cross sections / update models
  - DOUBLEHET available in SAMPLER
  - Manufacturing variations can be added
- Other Phase 1 exercises to be completed

Further outlook

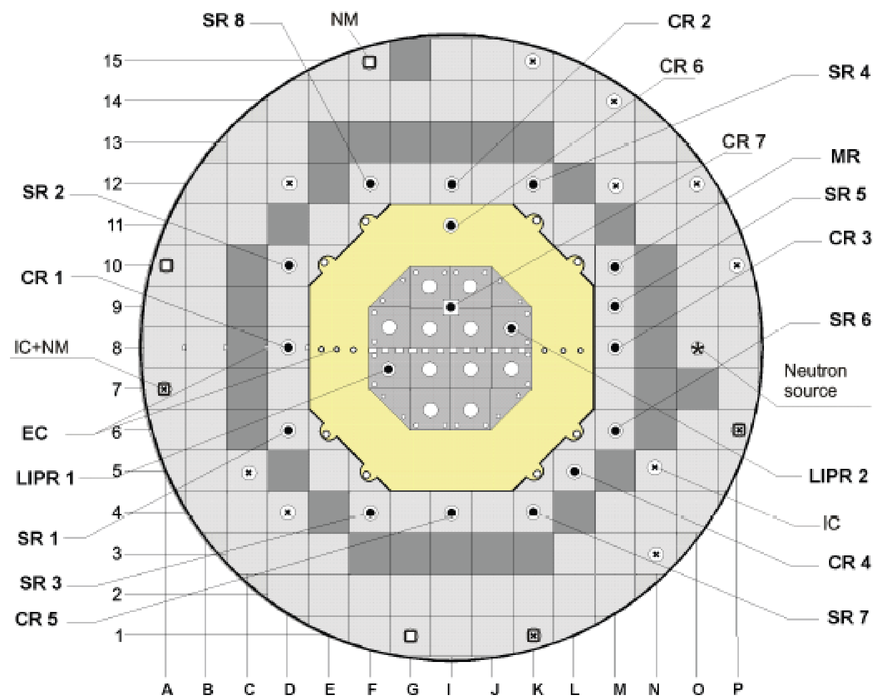
- Test cases for later phases to be finalised but will have to limit the scope (till end of 2019):
  - Depletion calculations
  - Full core coupled calculations
  - Limited transient cases



- Selected experimental results
  - ASTRA
  - VHTRC

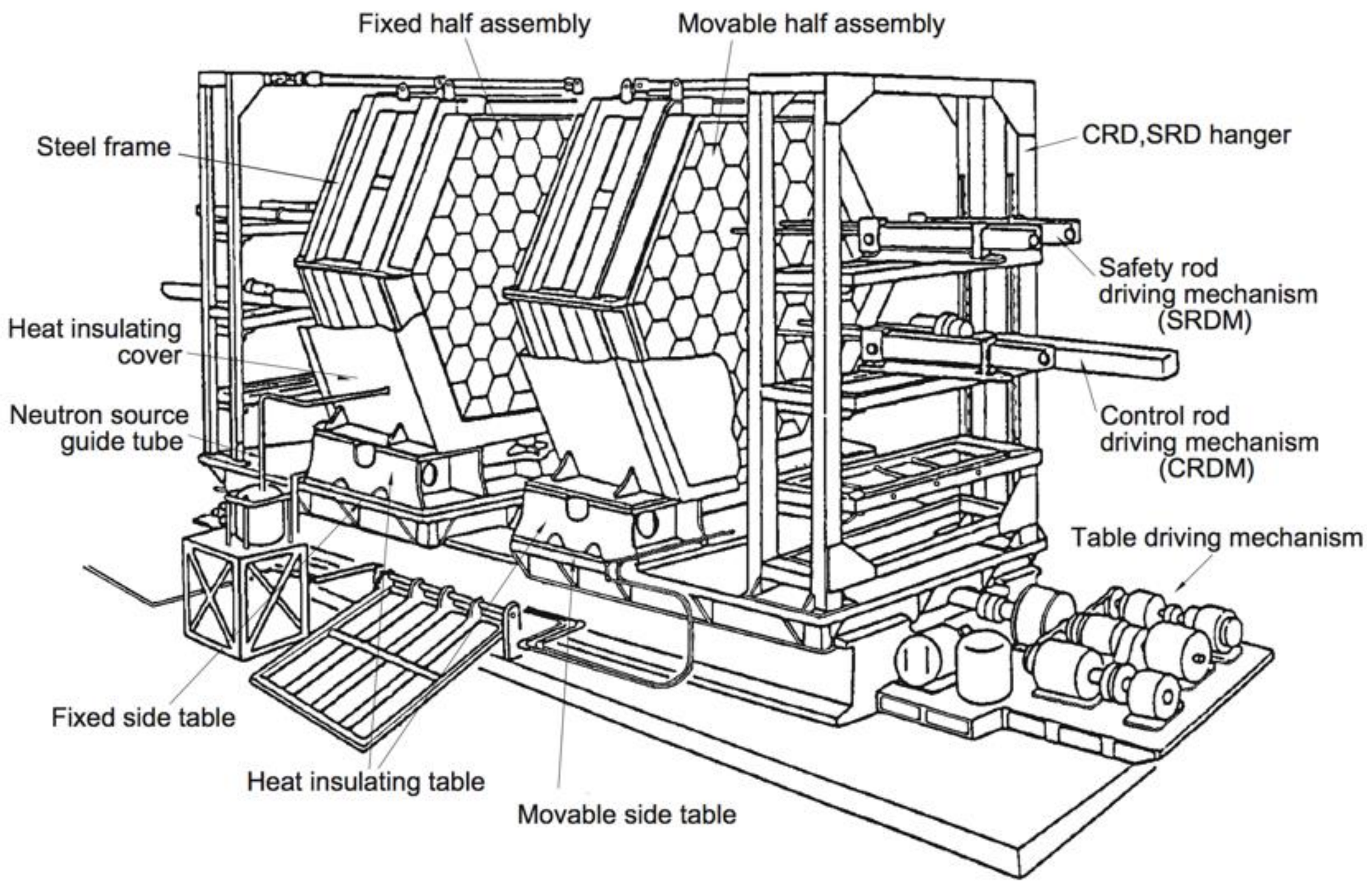
- TECDOC to be produced summarising all results (after 2019)

IEU-COMP-THERM-008 (ASTRA)



CR1, CR7	Control Rods
SR1, SR8	Safety Rods
MR1 (MR)	Manual Control Rod
LIPR1, LIPR2	Leave-In-Place Poison Rods
EC	Experimental Channels
IC	Ionization Chambers
NM	Neutron Meters (Neutron Counters)
Dark regions in the side reflector	Graphite Blocks with Plugs

VHTRC-GCR-EXP-001/CRIT-COEF



Concluding comment:

- Impact of the CRP is substantial !
  - More than 12 papers and publications within the last 18 months alone
  - 1xMSc and 1xPhD study completed
  - At least 2x PhD projects direct coupled to the project – several others related
- Several codes are being further developed to be able to do this work
- Knowledge gained will be beneficial for design and safety analysis of future HTGRs

Power and flux shaping

Number of passes…

Comparison study between different pass cases in VSOP HTR models

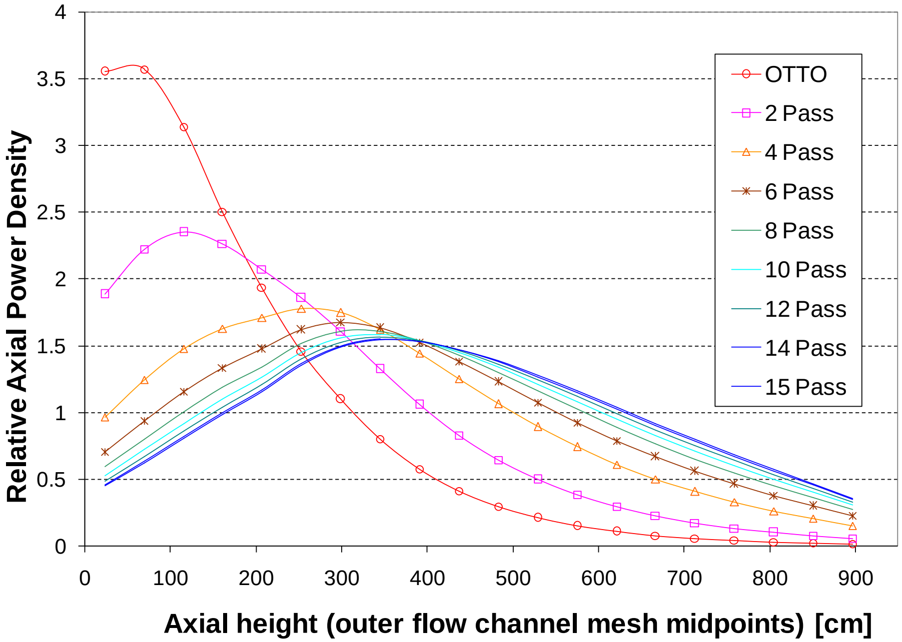
Wilna Geringer

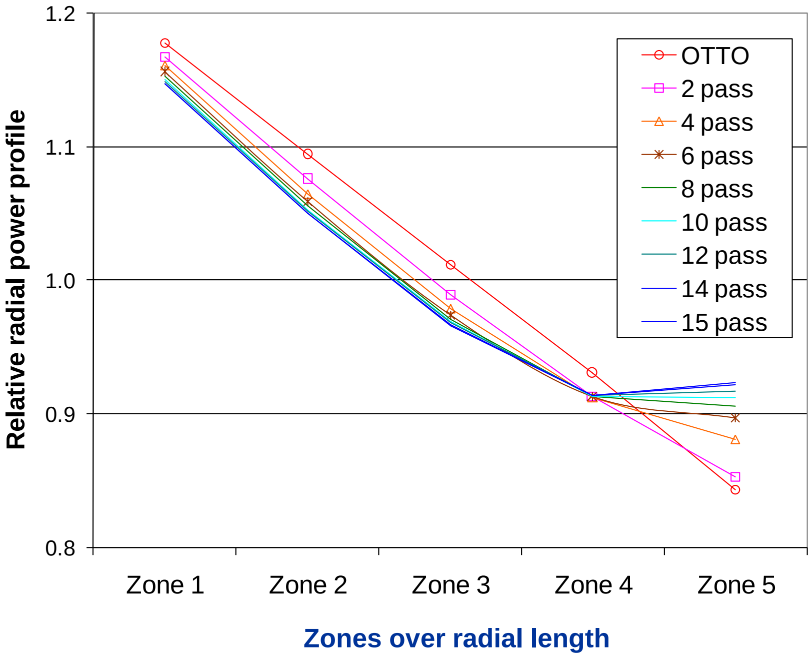
Frederik Reitsma

HTR 2010 – Energy for Industry
5th International Conference on High Temperature Reactor Technology

HTR Module – Power profile results

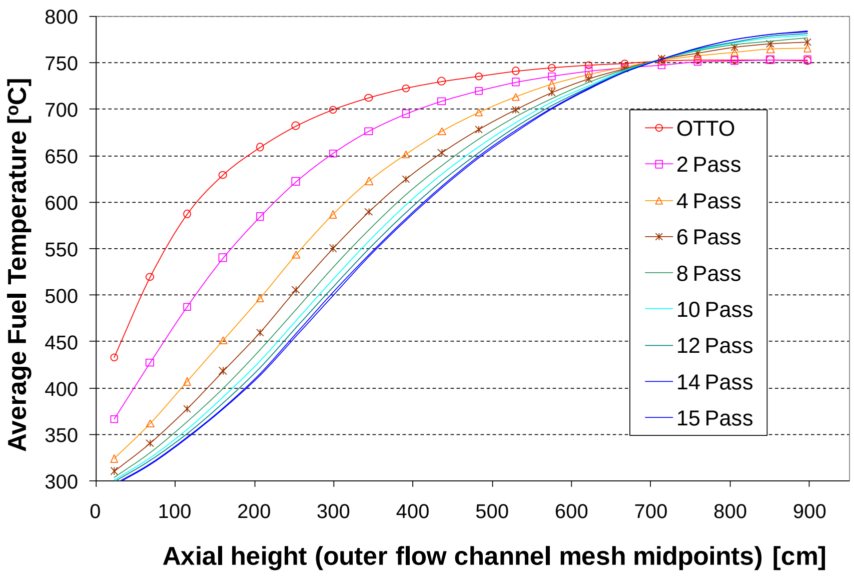
- For an optimal core design it is required to achieve a flattened power distribution over the core.
- Fresher fuel mixture causes peaking at the top of the core (in the low pass numbers)
- Radial power profile impact is limited (small benefit for multiple passes)

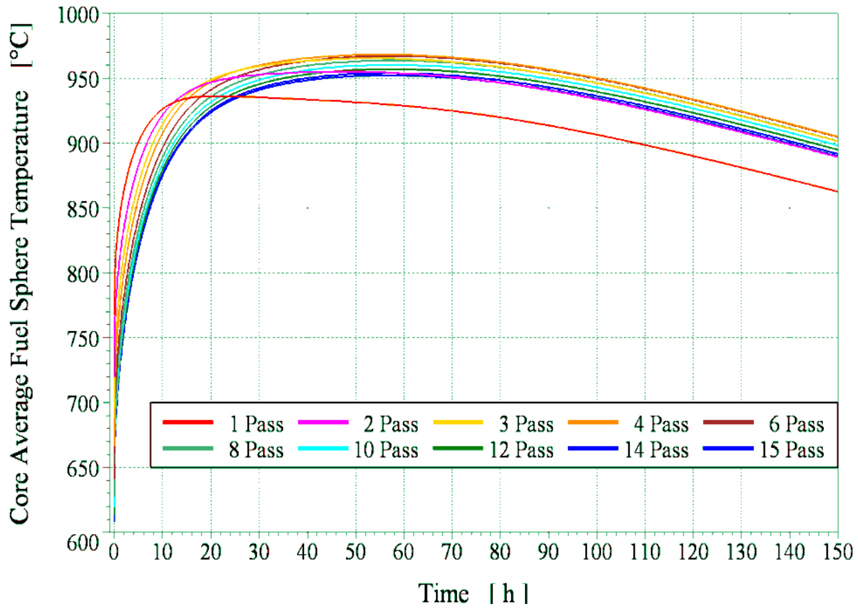




HTR Module – Fuel temperature results

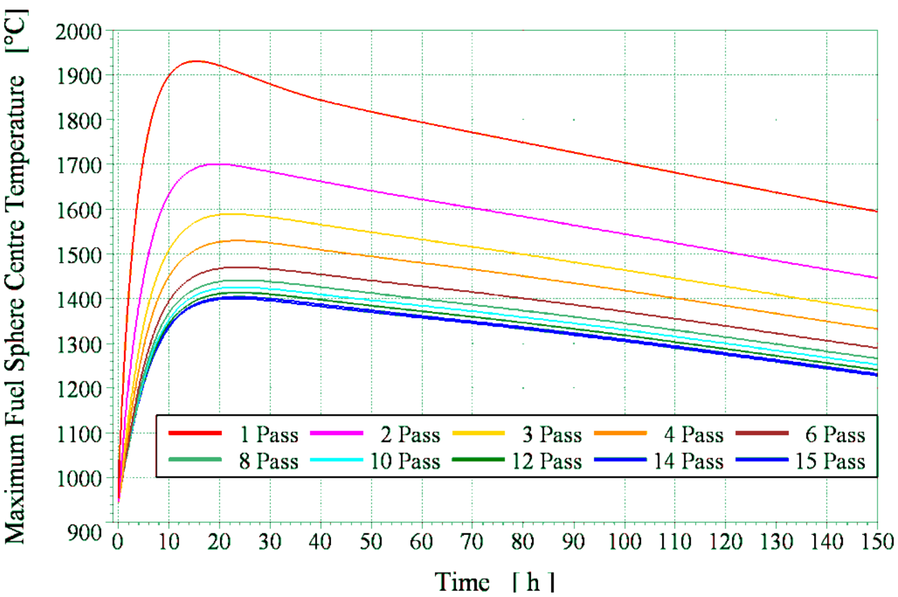
- Temperatures must remain under 1600ºC for normal and accident conditions.
- On average the OTTO cycle has the highest fuel temperature (756ºC). The opposite is true for DLOFC conditions.
- The maximum fuel temperature increases with the increase in number of passes. A difference of 72ºC between the highest and the lowest are obtained.

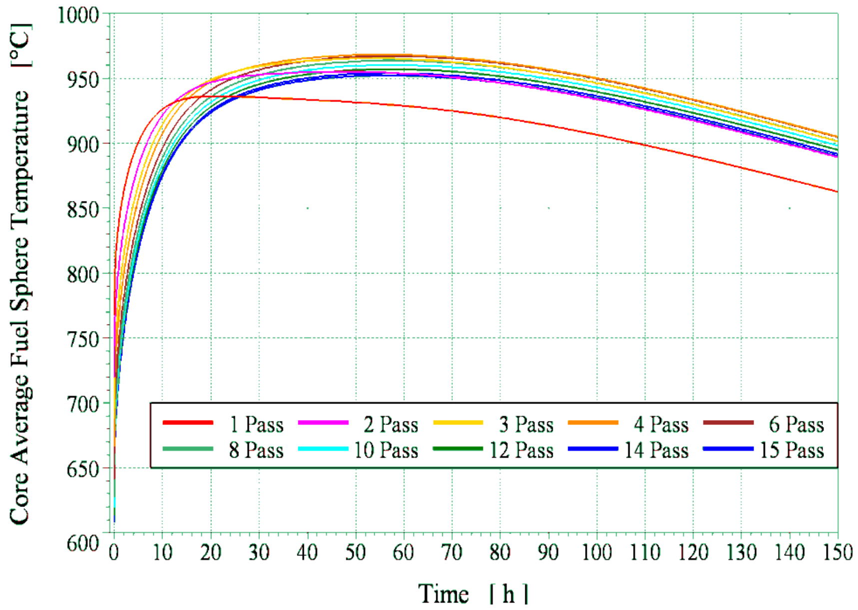




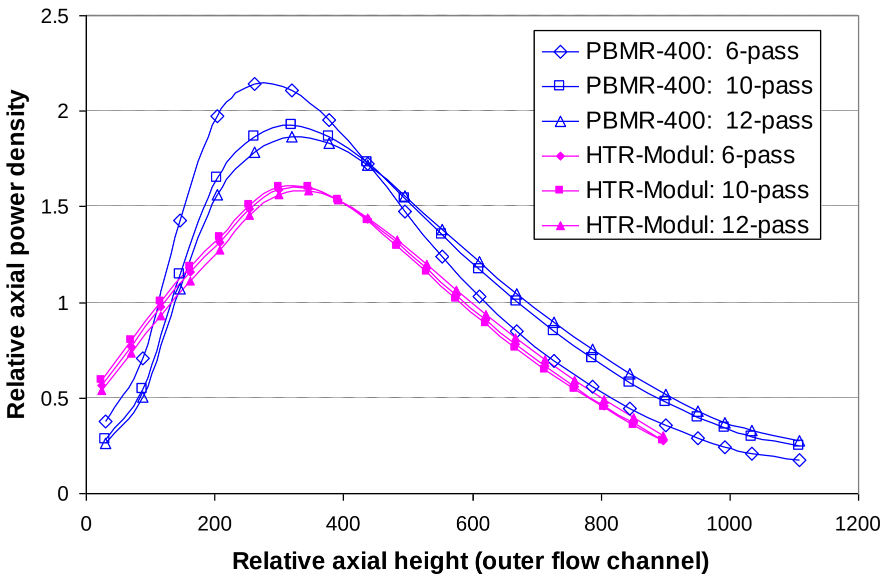
HTR Module – Fuel temperature results

- Temperatures must remain under 1600ºC for normal and accident conditions.
- On average the OTTO cycle has the highest fuel temperature (756ºC). The opposite is true for DLOFC conditions.
- The maximum fuel temperature increases with the increase in number of passes. A difference of 72ºC between the highest and the lowest are obtained.
- For DLOFC conditions the OTTO-pass case and the two-pass case goes above specified conditions and temperatures over and at 1900ºC and 1700ºC are achieved respectively.





PBMR 400MW – Multi-pass results



Overall performance:

- Small increase in the average discharge burnup with the increased number of passes.
- The peaking of the PBMR 400MW is larger than for the HTR-Module.
- The power and temperature behaviour of the PBMR 400 MW design are similar to those for the HTR-Module, but due to the higher peaking and the effect of the control rod design it benefit even more from additional passes.

Case	Fuel residence time [days]	Av. discharge burnup [MWD/t]	Neutron leakage
6	956	94023	14.8%
10	972	95582	14.6%
12	975	95952	14.6%

Case	Max. Fuel Power [kW]	Av. Fuel Temp. [^{o}C]	Max. Fuel Temp. [^{o}C]
6	3.00	884	1104
10	2.86	871	1107
12	2.83	868	1109

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Graphite fluence behaviour and core structure lifetime evaluation

GEOMETRIC LAYOUT OPTIMISATION OF GRAPHITE REFLECTOR COMPONENTS

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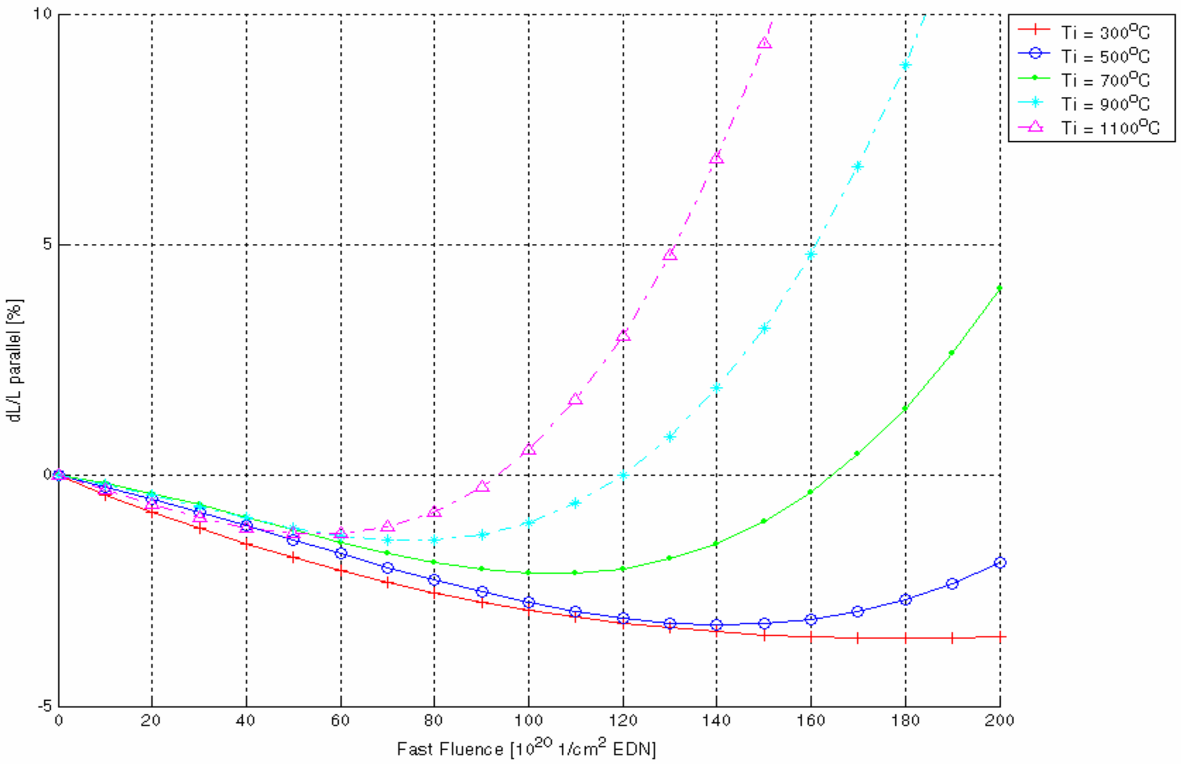
& Michael Hindley
makke@mweb.co.za
Formerly Pebble Bed Modular Reactor (Pty) Ltd.
Centurion, South Africa.

Presented By
F Reitsma
Pebble Bed Modular Reactor (Pty) Ltd.
Centurion, South Africa.

Purpose of Analyses

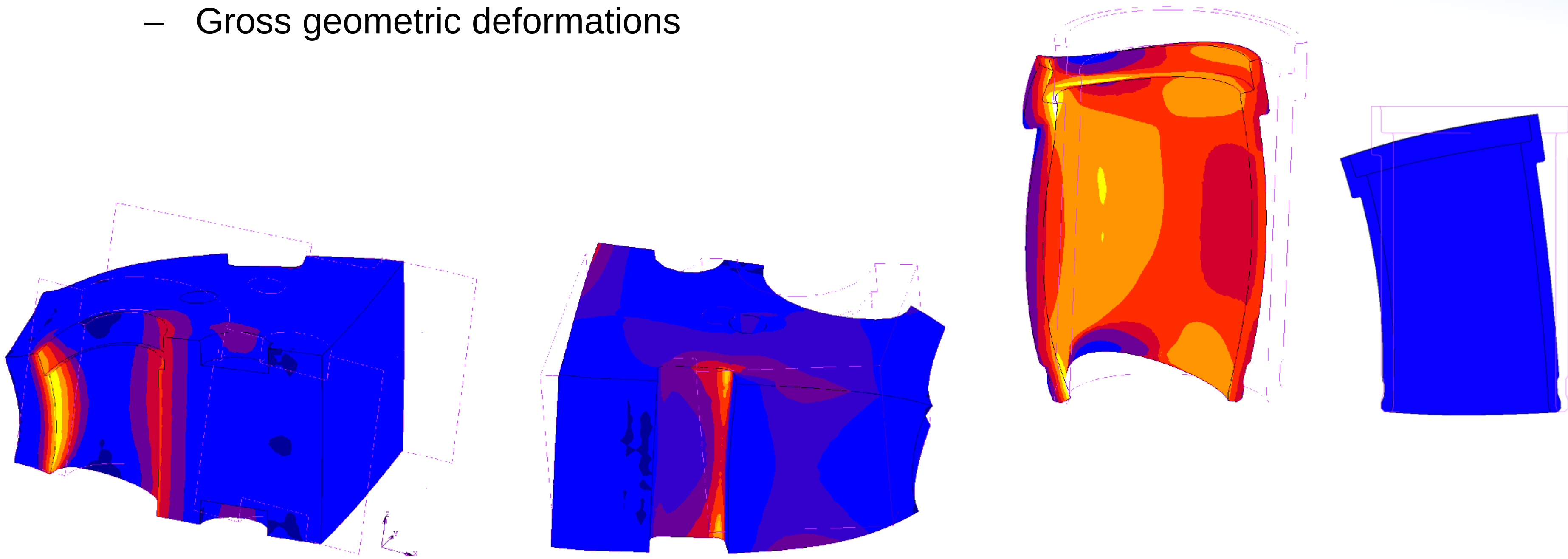
- The aim of the graphite analyses is to assess the life that can be expected from the replaceable reflector components
- Graphite core structures are subject to extreme thermal loading
- Fast neutron fluence causes material property changes that lead to extremely non-linear material behavior



- Material properties change with temperature and neutron fluence
- The graphite shrinks and swells

Behavior

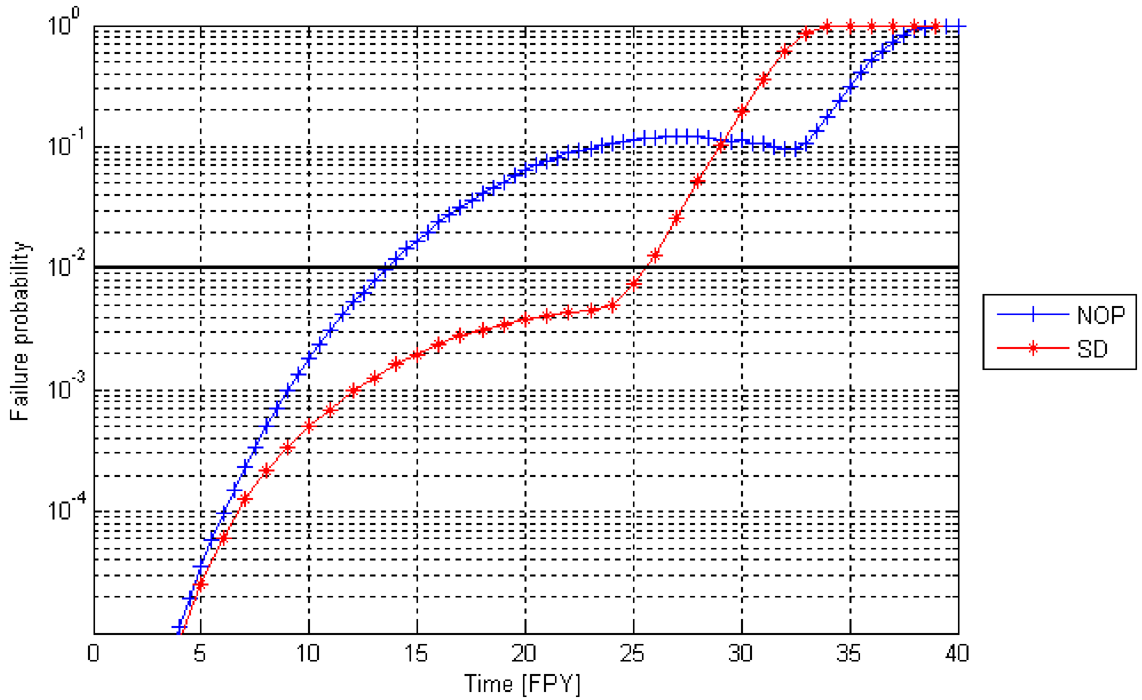
- The load induced shrinkage and swelling can cause:
  - High internal stresses
  - Gross geometric deformations

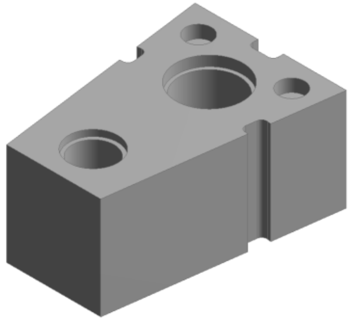


- We ultimately have to determine the safe stress levels and lifetimes of the core components

Baseline Results

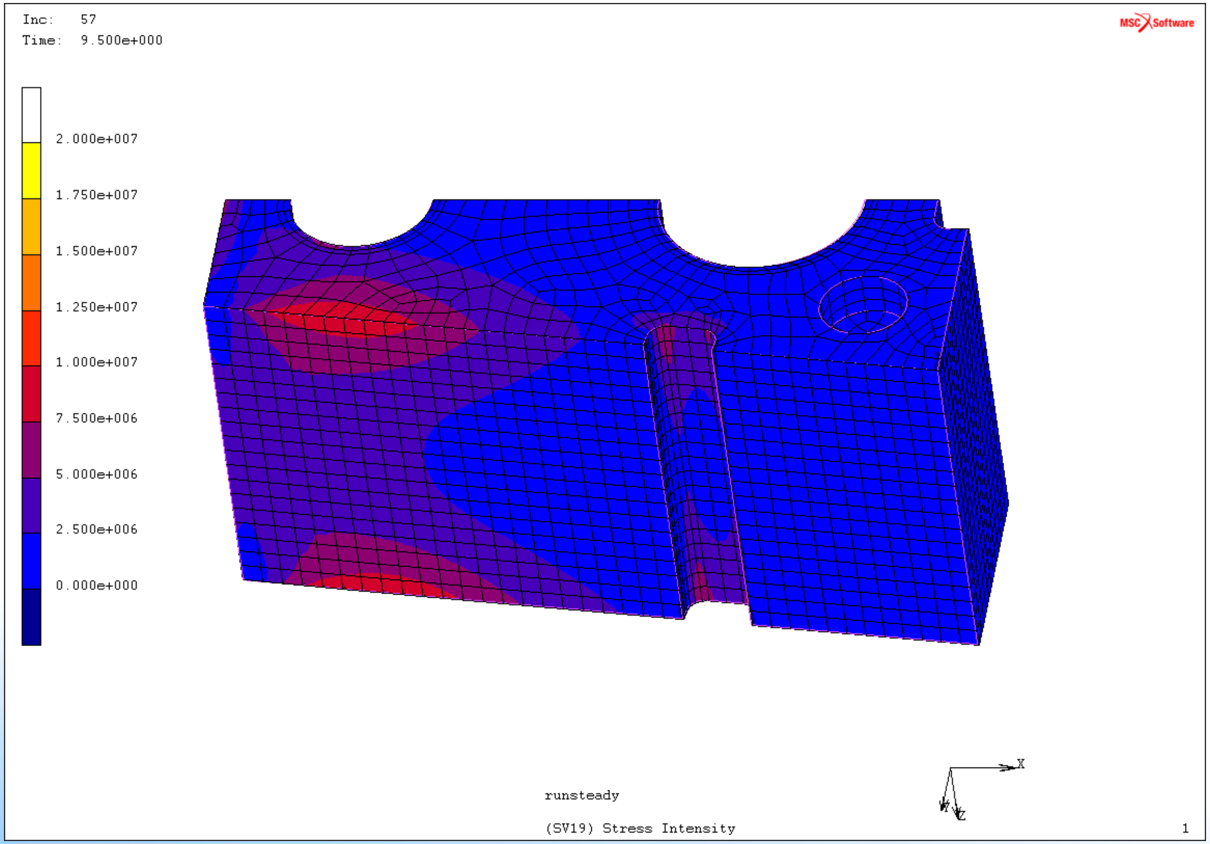
- Predicted Failure probability of baseline block

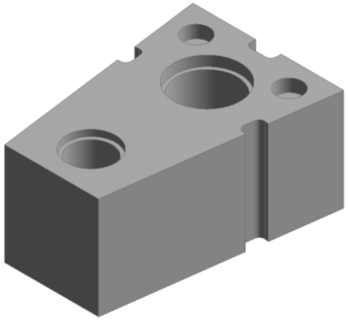




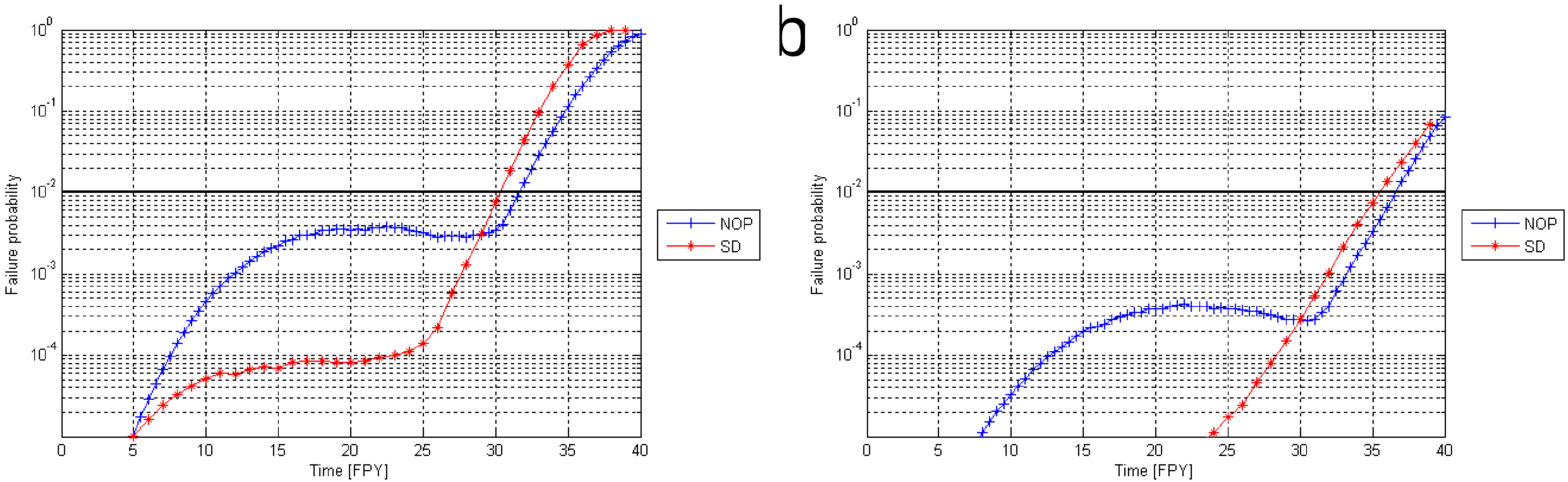
Baseline Results

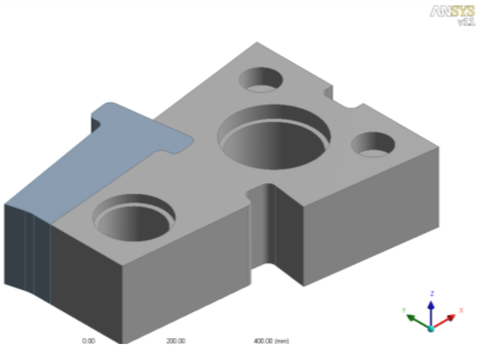
- Stress intensity (Maximum deformation energy MDE) in baseline block at predicted end of life





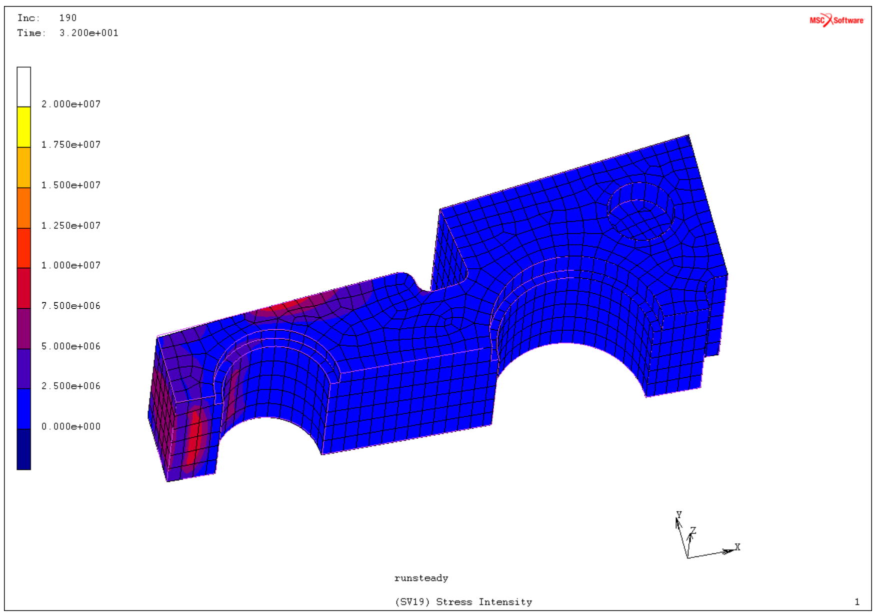
Proposed Design Results

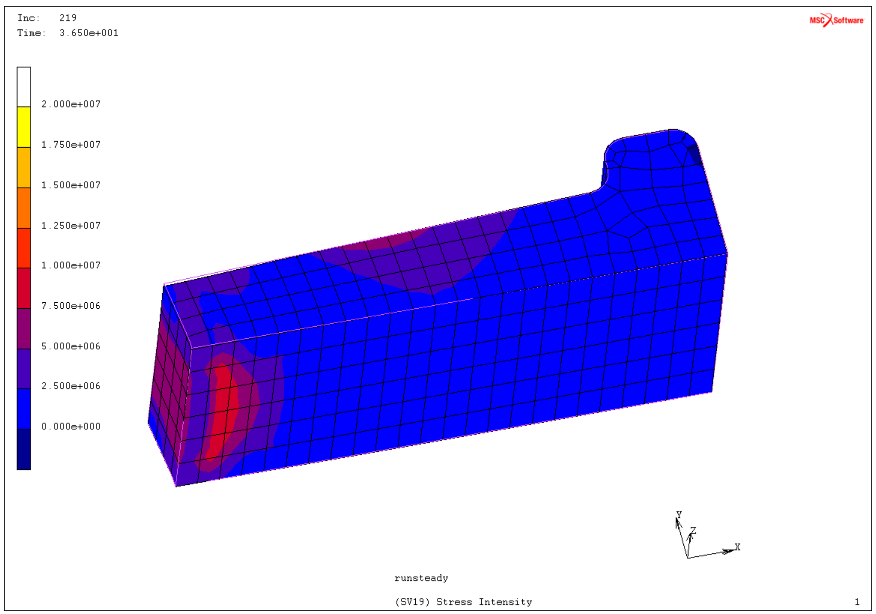


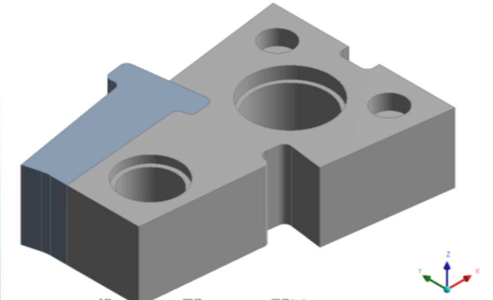


Proposed Design Results

- Stress intensity (MDE) in proposed blocks at predicted end of life

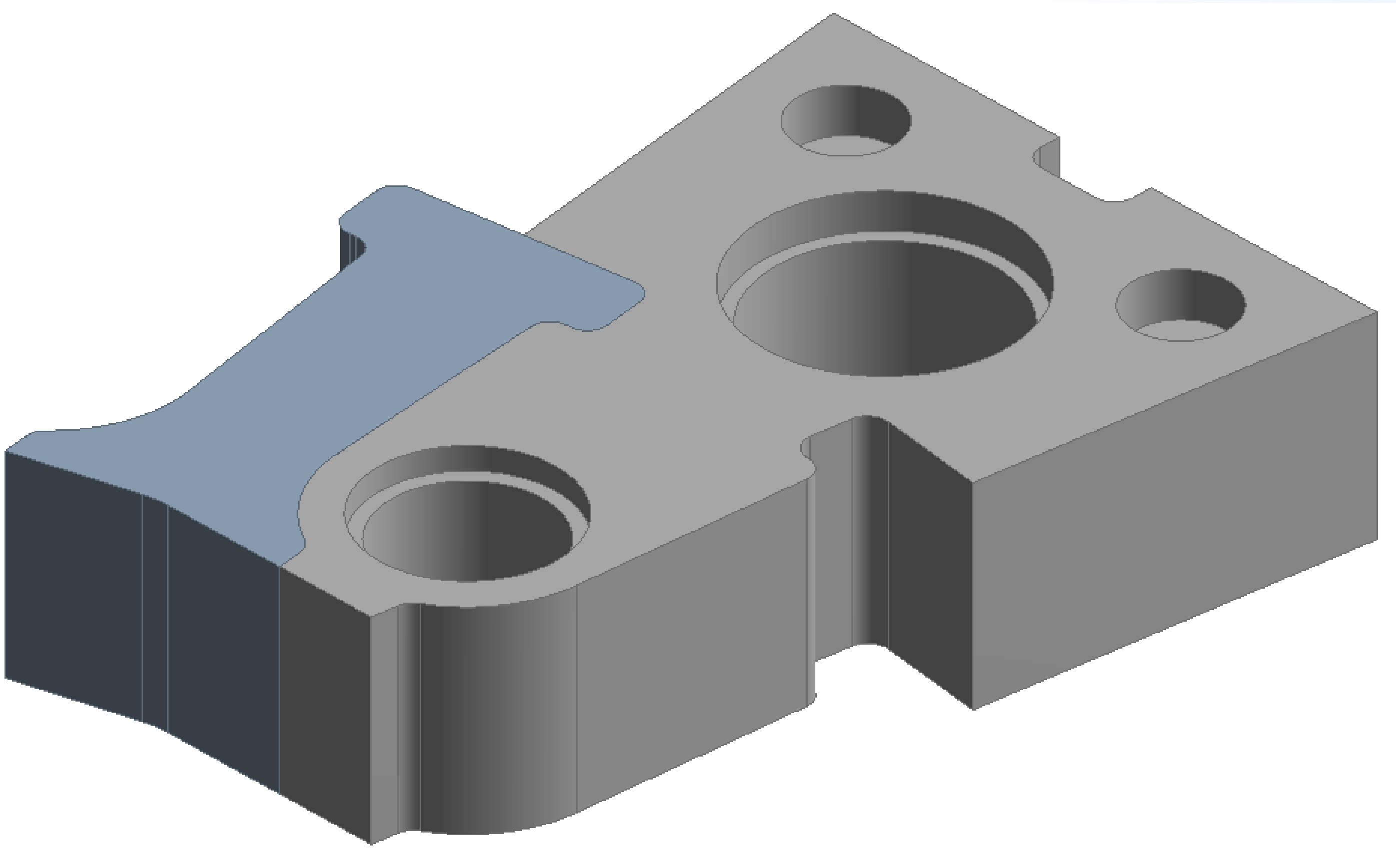




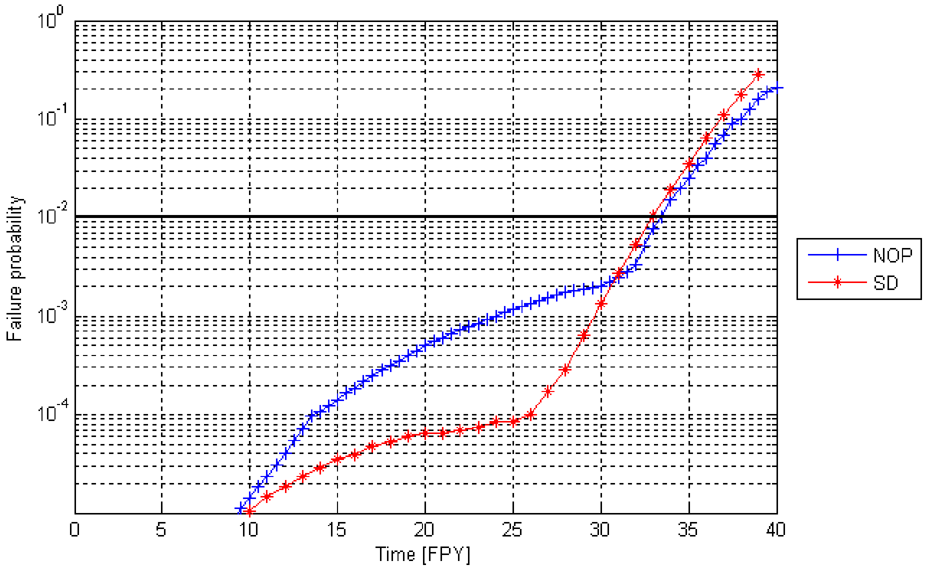


Further Improvements

- Using topology optimization the following further Improvement can be obtained.

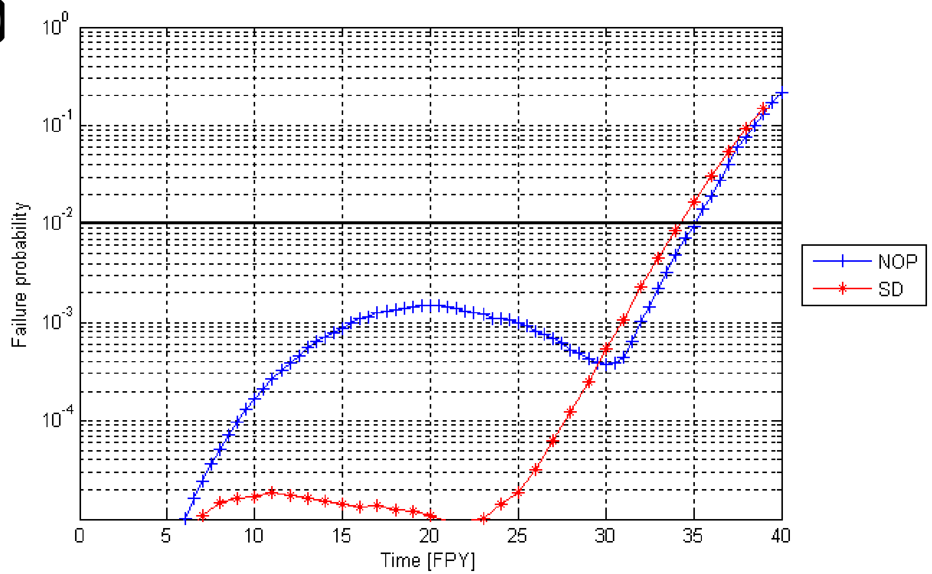


Results

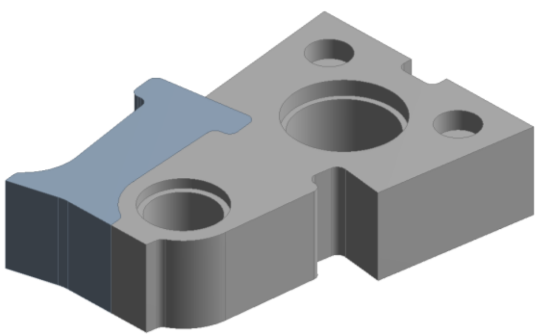


Failure probability
10^0
10^-1
10^-2
10^-3
10^-4
0 5 10 15 20 25 30 35 40
Time [FPY]
NOP
SD

b



Failure probability
10^0
10^-1
10^-2
10^-3
10^-4
0 5 10 15 20 25 30 35 40
Time [FPY]
NOP
SD



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Graphite Dust

EXAMINATION OF DUST IN AVR PIPE COMPONENTS

Collecting loose dust

- The expected loose dust should be collected after removal of the first plug. ….. Then the pipes were turned so that the open end was located at the bottom and the pipe walls were knocked with a hammer for some minutes. The sample collectors were removed after a settling time of more than one hour. ….. The amount of loose dust material was negligible and was nearly invisible between the swarf. … it was not possible to isolate the small amounts (approximately below 1 mg) of loose dust from the swarf therefore no further investigations could be made.

Some more results

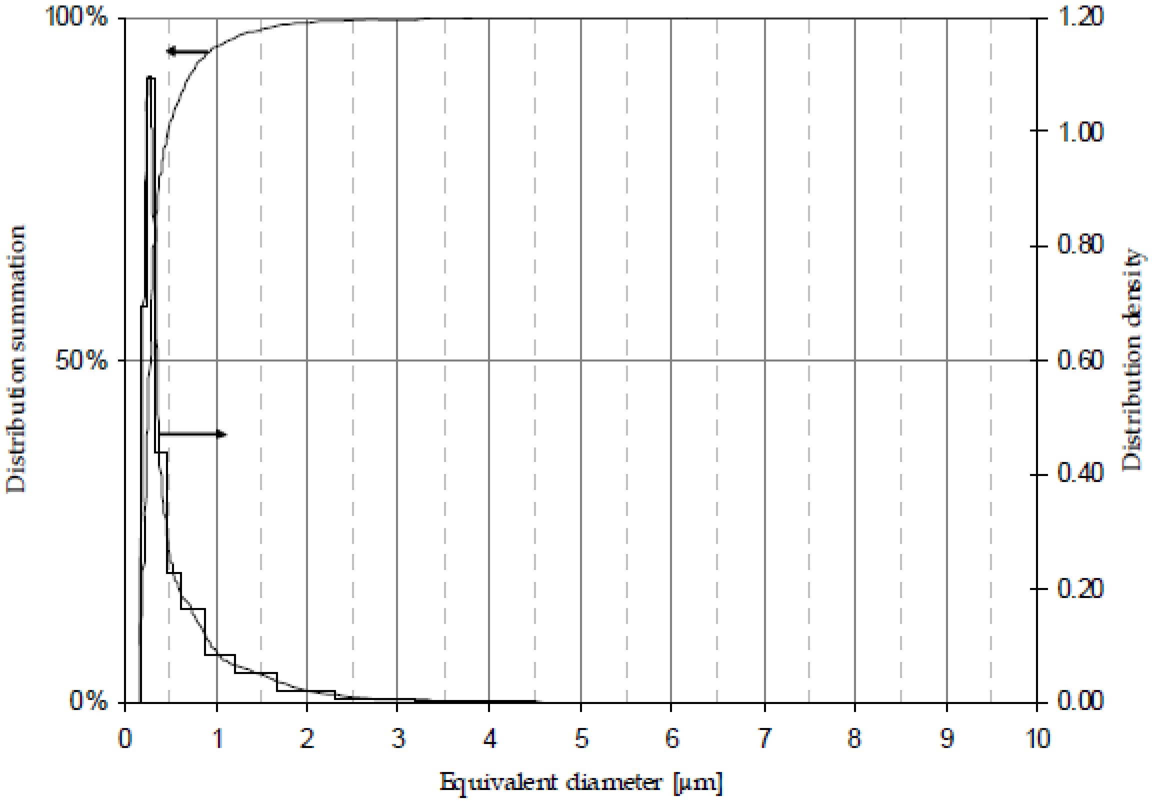

Figure 19: Number weighted distribution of removed de-agglomerated dust material from E7

Scraping the dust layer from a mounted segment

Main aspects

- Dust itself is not an issue
- Dust can serve as a means of transport of fission products that adhere to the dust surfaces
- Dust sources
  - Mechanical
  - Chemical
- Deagglomerated dust distribution (<1 micron) not consistent with mechanical wear and tear – more consistent with chemical forming
- AVR experience may not be consistent with what can be expected
  - Water ingress
  - Oil ingress
  - Small mechanical loadings

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“fresh fuel can cluster together and cause huge power peaking or hot spots”

.. Not true

Papers by PBMR and INL

(not the reason for high temperatures in AVR.. Not impossible to quantify… not a safety concern due to margins)

INVESTIGATION OF THE POWER PEAKING IN THE PBMR PEBBLE-BED REACTOR

Frederik Reitsma and Wessel R. Joubert
Nuclear Engineering Analysis (NEA), PBMR (Pty Ltd)

Abderrafi M. Ougouag and Hans D Gougar
Idaho National Laboratory, Idaho Falls

Results & Conclusions

- Results
  - Large safety limits
    - increase in power per fuel sphere still far below set limit ☺
    - 2.8 kW reference far below 4.5kW/fs i.e. PBMR set limit
  - Displacement
    - 3.0 kW per fuel sphere estimated for a cluster of 20 fresh fuel elements
  - Clustering (35cm x 30cm ring on power peak)
    - large increase in volumentric power density: 11.3 MW/m^3 -> 18.7 MW/m^3
    - only 3.1 kW per fuel sphere for large cluster
  - Small increase in maximum fuel temperatures ☺
- Why?
  - flux and spectrum dominated by environment
  - in a cluster all the FS's is now contributing equally – small local variation in powers
- Simple approach used show no severe effects due to displacement of clustering

AVR – “hot spots” and high measured temperatures

The Re-evaluation Of The AVR
Melt-wire Experiment Using Modern Methods With Specific Focus On Bounding The Bypass Flow Effects.

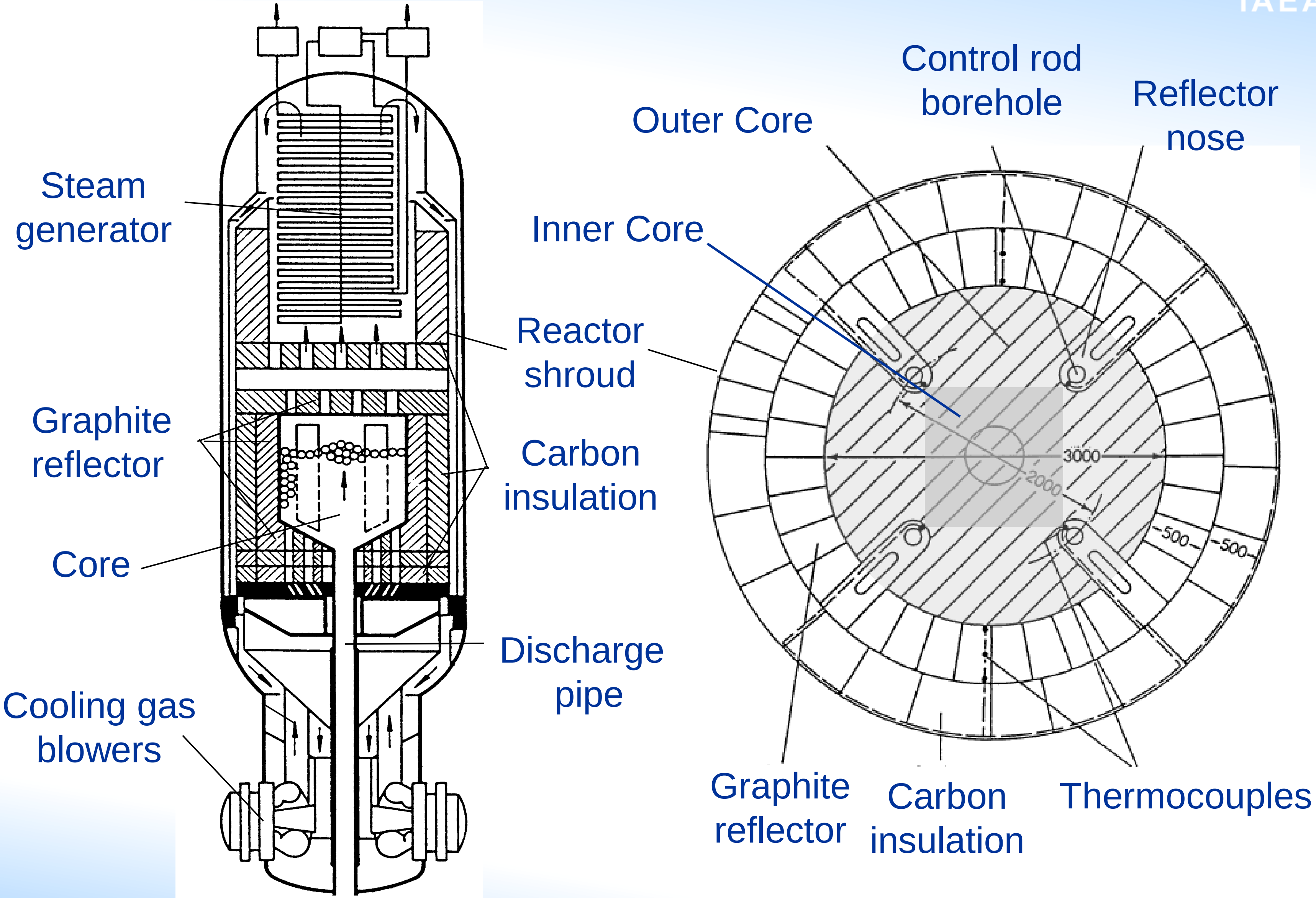
CF Viljoen, S Sen, F Reitsma, O Ubbink, P Pohl, H Barnert

4th International Topical Meeting on High Temperature Reactor Technology

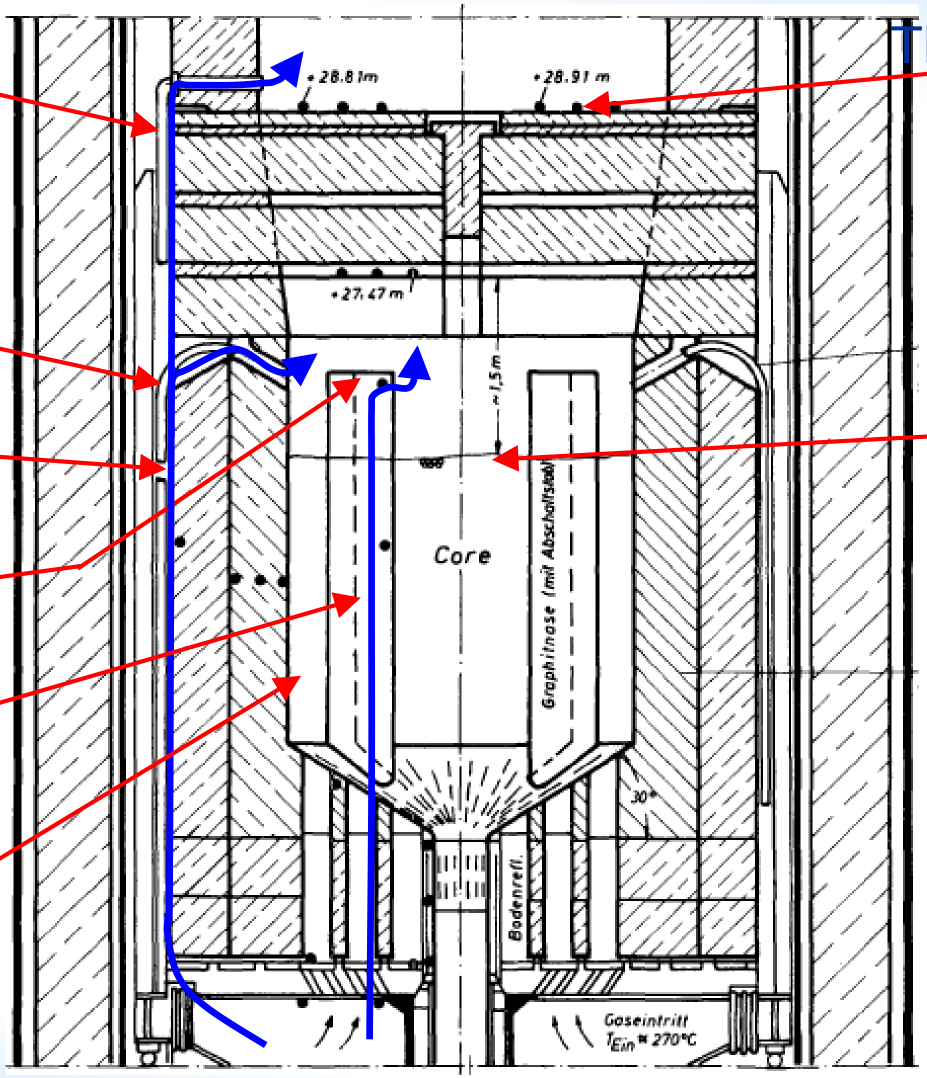
Background

- AVR (Arbeitsgemeinschaft Versuchsreaktor)
  - Research reactor
  - Test bench for different pebble fuel types
- Operated for 21 Years
- Bypass flows were not included in calculations
  - only considered after melt-wire tests in 1988

AVR layout


Steam generator
Graphite reflector
Core
Cooling gas blowers
Reactor shroud
Carbon insulation
Discharge pipe
Outer Core
Control rod borehole
Reflector nose
Inner Core
3000
2000
500
500
Graphite reflector
Carbon insulation
Thermocouples

Core layout & measurements



Bypass pipes

Fuelling lines

Annular gap

Top plug

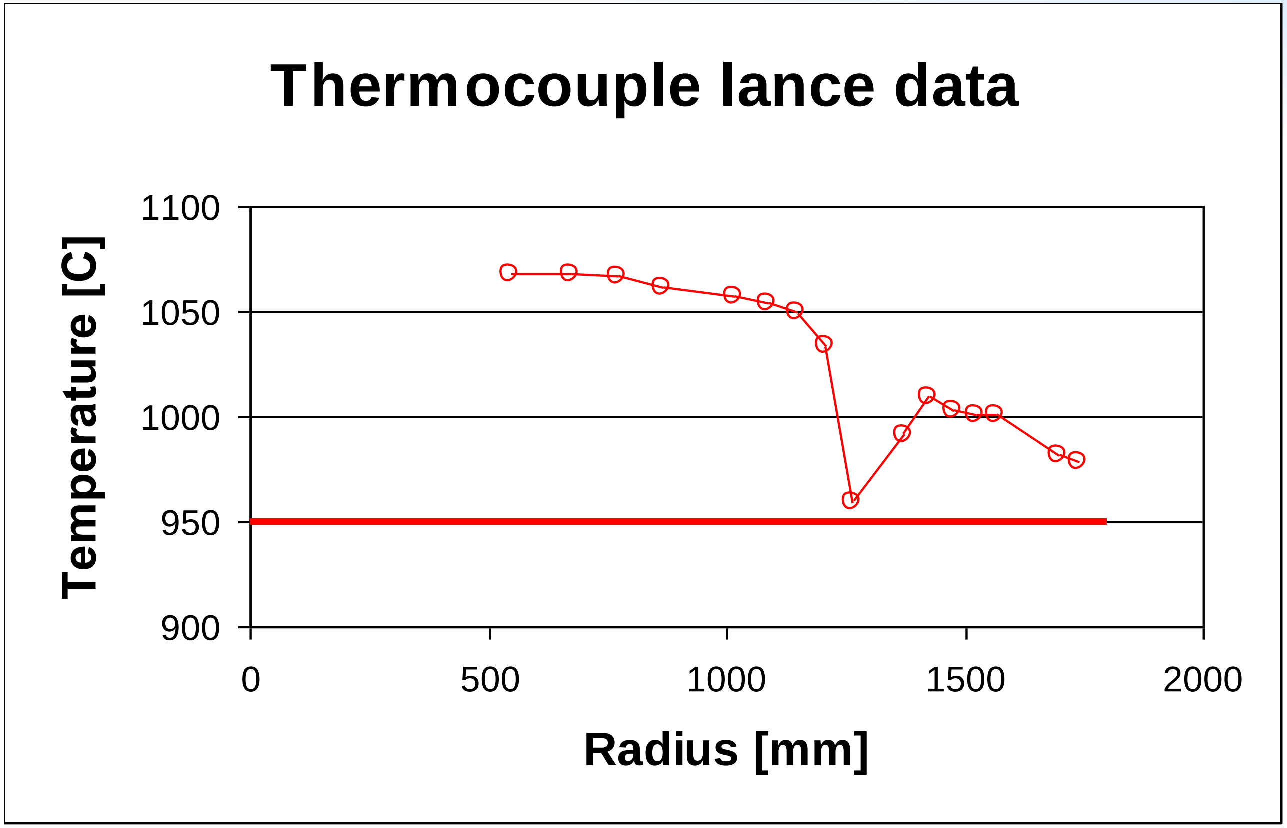
Control rod nose

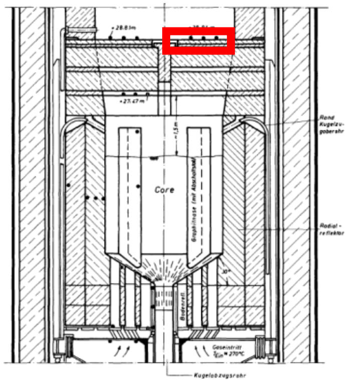
Wall channeling

Thermocouple lance position

Melt-wire measurement position

Outlet Temperature Higher & Uncertain

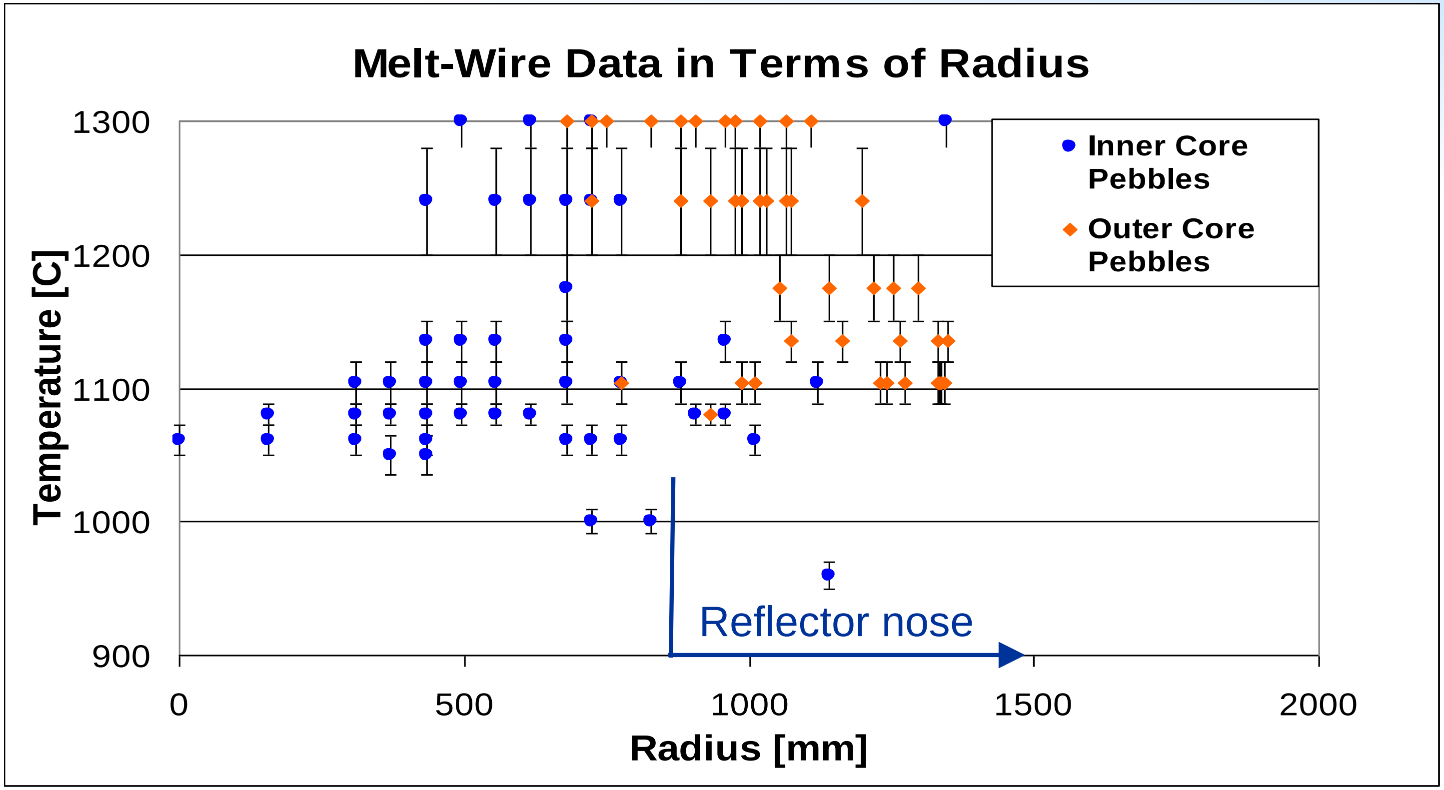


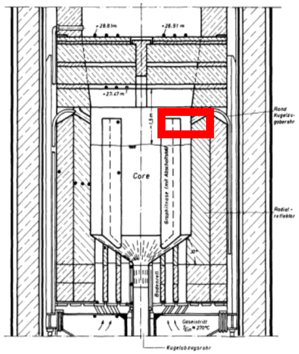


Average value =1024°C
Uncertainty at R=1300mm

AVR-Meltwires

Interpreted Melt-Wire Data

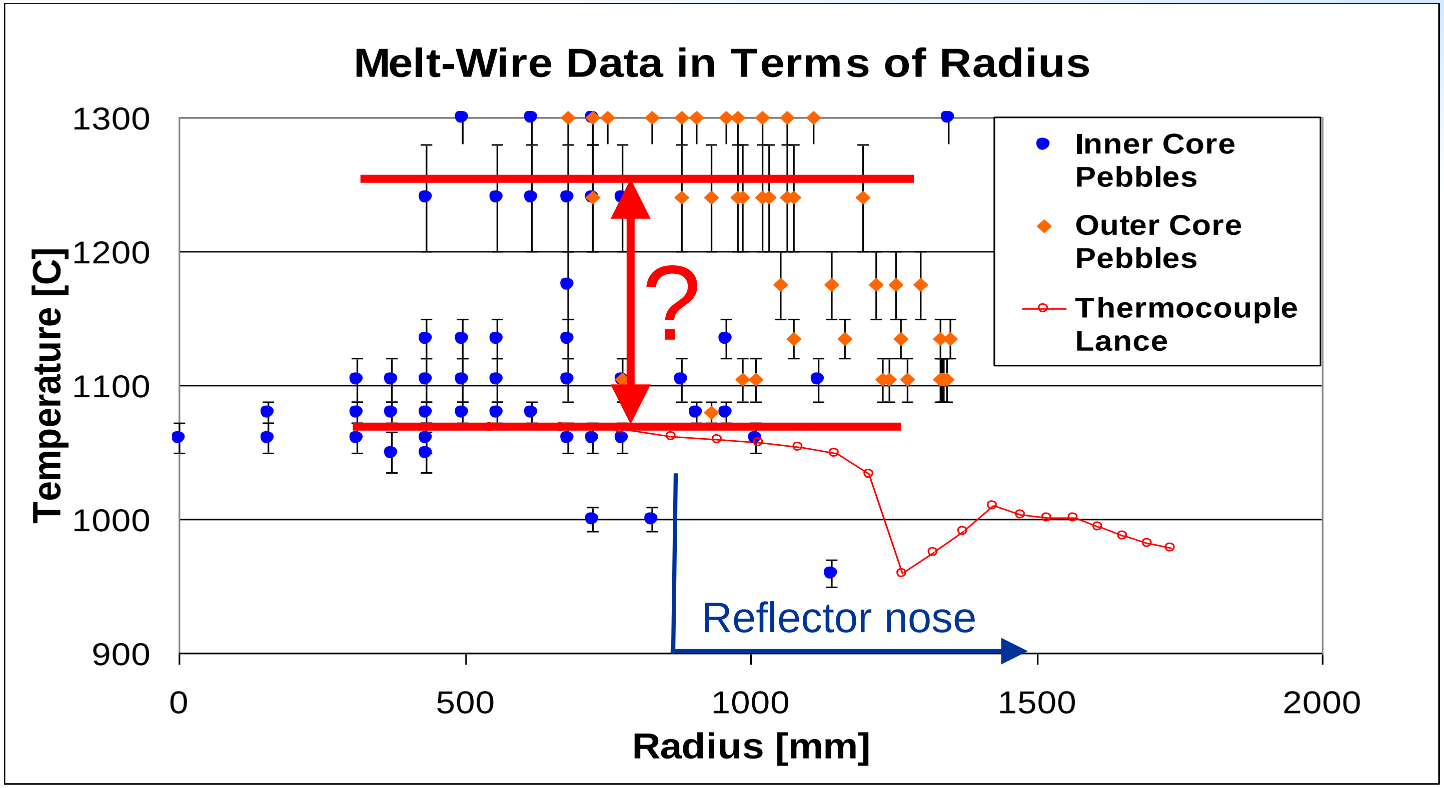


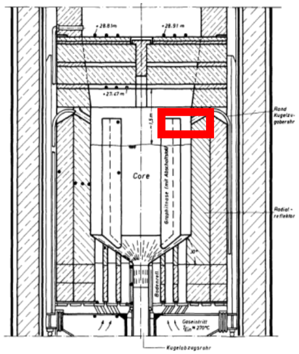


Average value >1136°C

Little variation in inner core

Interpreted Melt-Wire Data



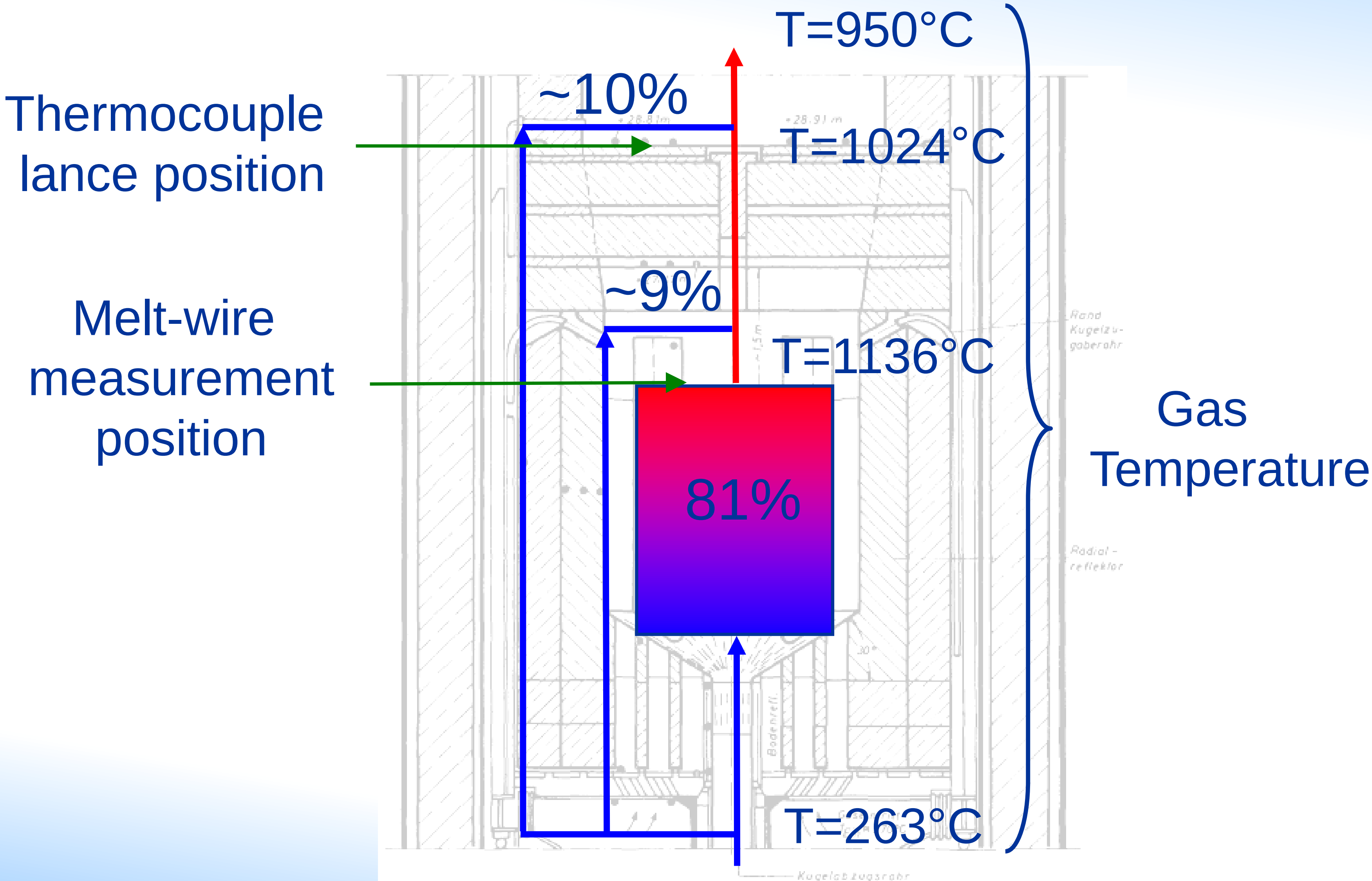


Average value >1136°C

Little variation in inner core

Difference between meltwire data and lance

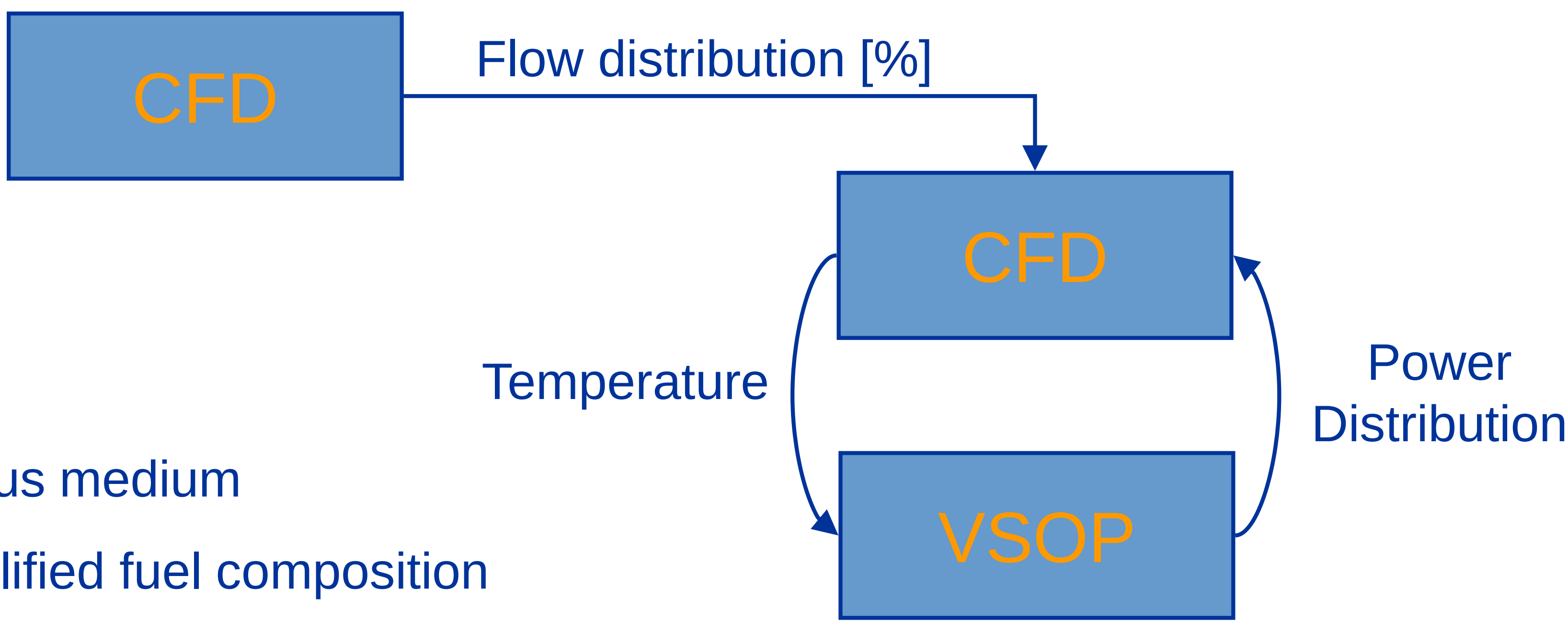
Estimation of bypass flows from measurements



Models

Detail Geometry

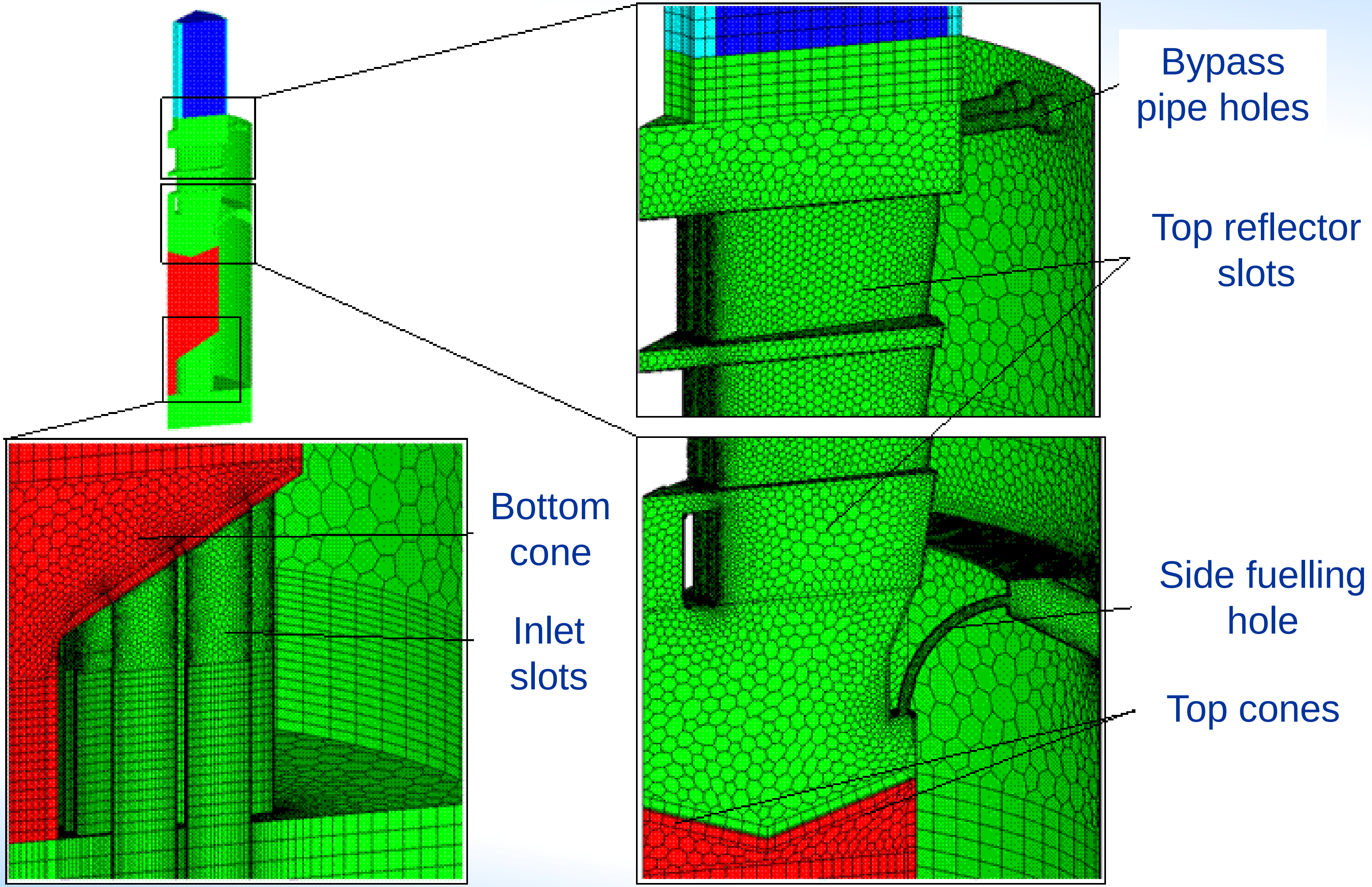
Simplified Geometry



Porous medium

Simplified fuel composition

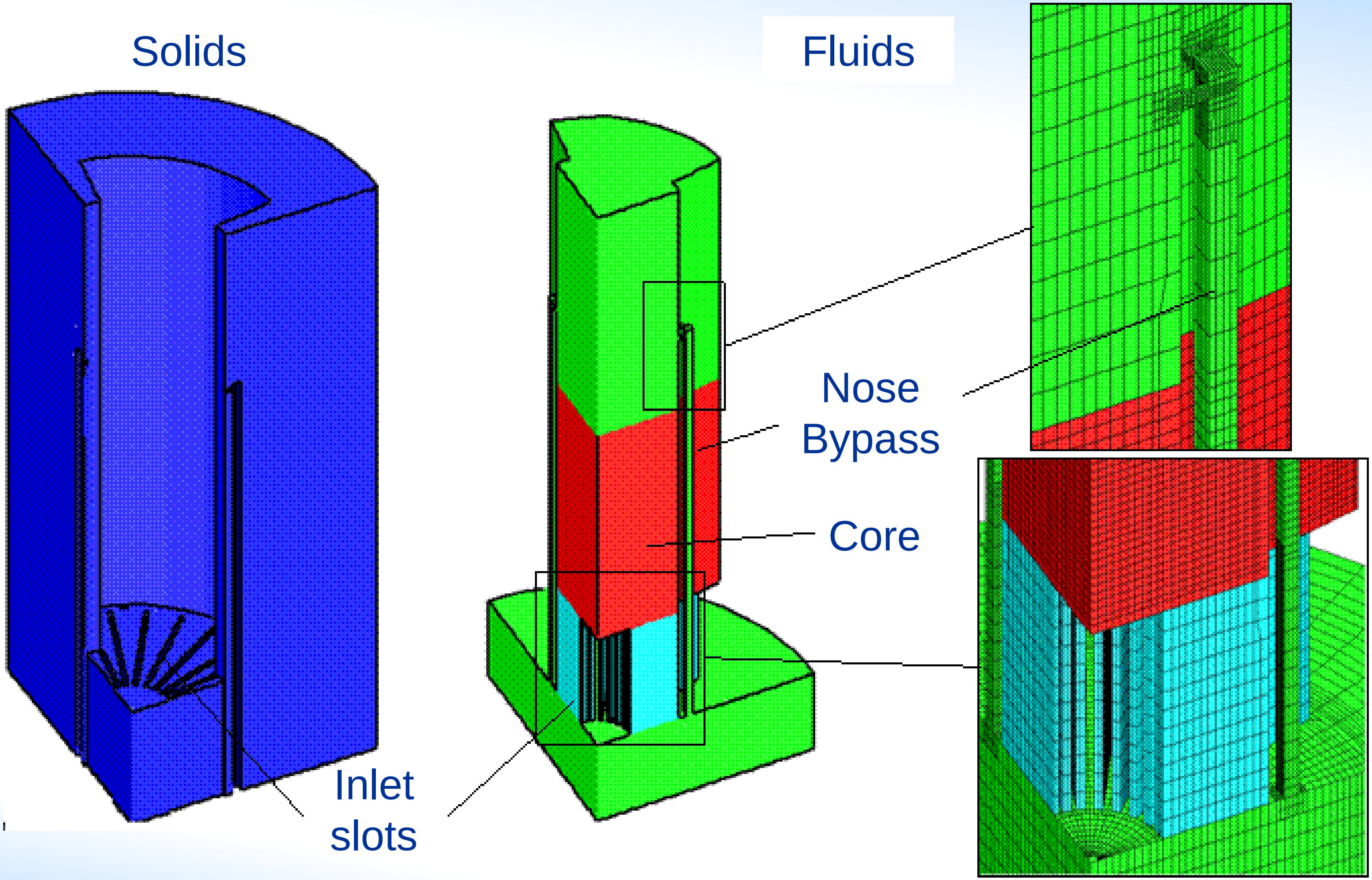
Detail Flow Model



Detail flow model: Flow Distribution

Annular Gap [mm]	Flow Distribution [%]		
	Core	Control Rod Boring	Annulus
2.0	91.4	7.3	1.3
5.6	82.0	6.8	11.2
7.0	78.3	6.5	15.2

Simplified Coupled CFD Model



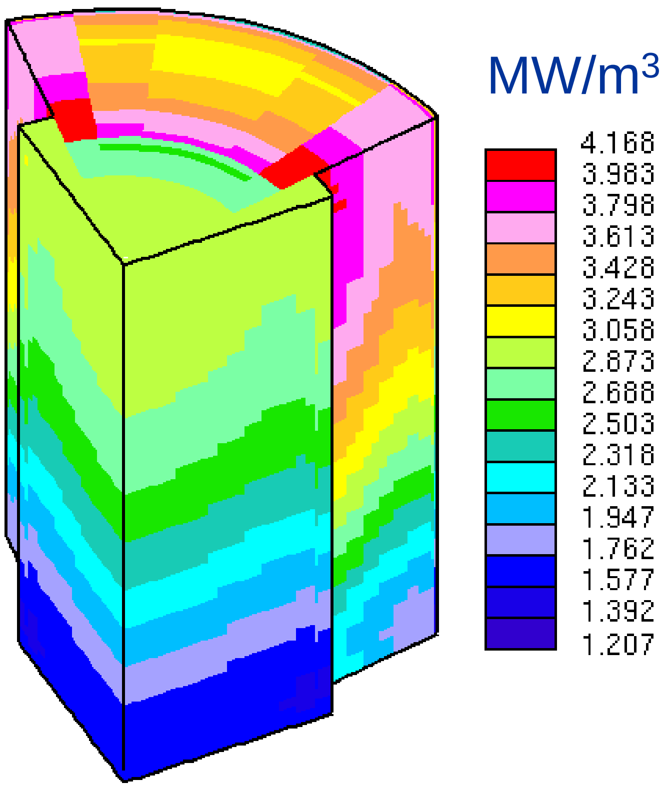
Outlet Gas Temperature Increase due to Bypass Flow

Case	Control Rod Bypass [%]	Annular bypass [%]	Wall channeling	Max Gas Temperature [°C]
1	0	0	No	1058
2	7	0	No	1102
3	7	10	No	1194
4	7	10	Yes	1209

Typical Core Conditions with Bypass Flow

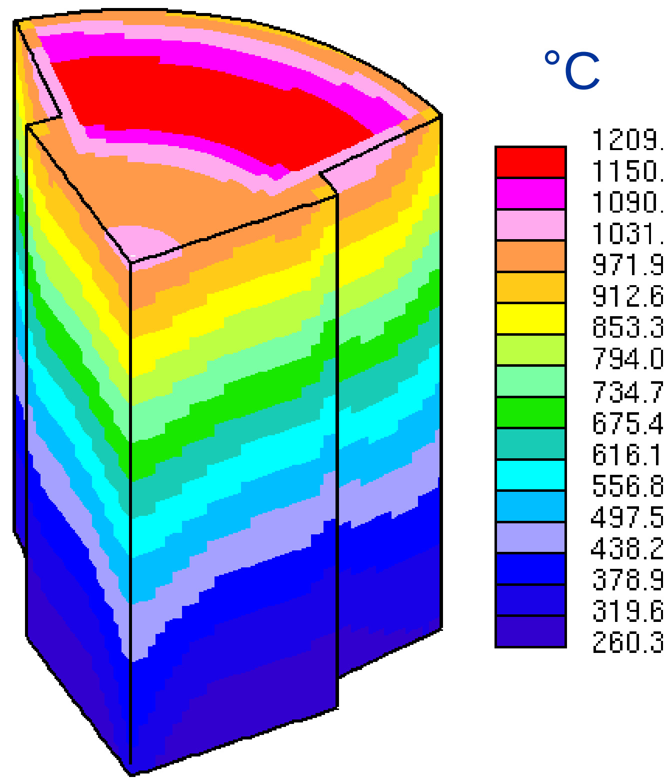
VSOP

Power Distribution



CFD

Gas Temperature



Impact of fuel loading on gas temperature

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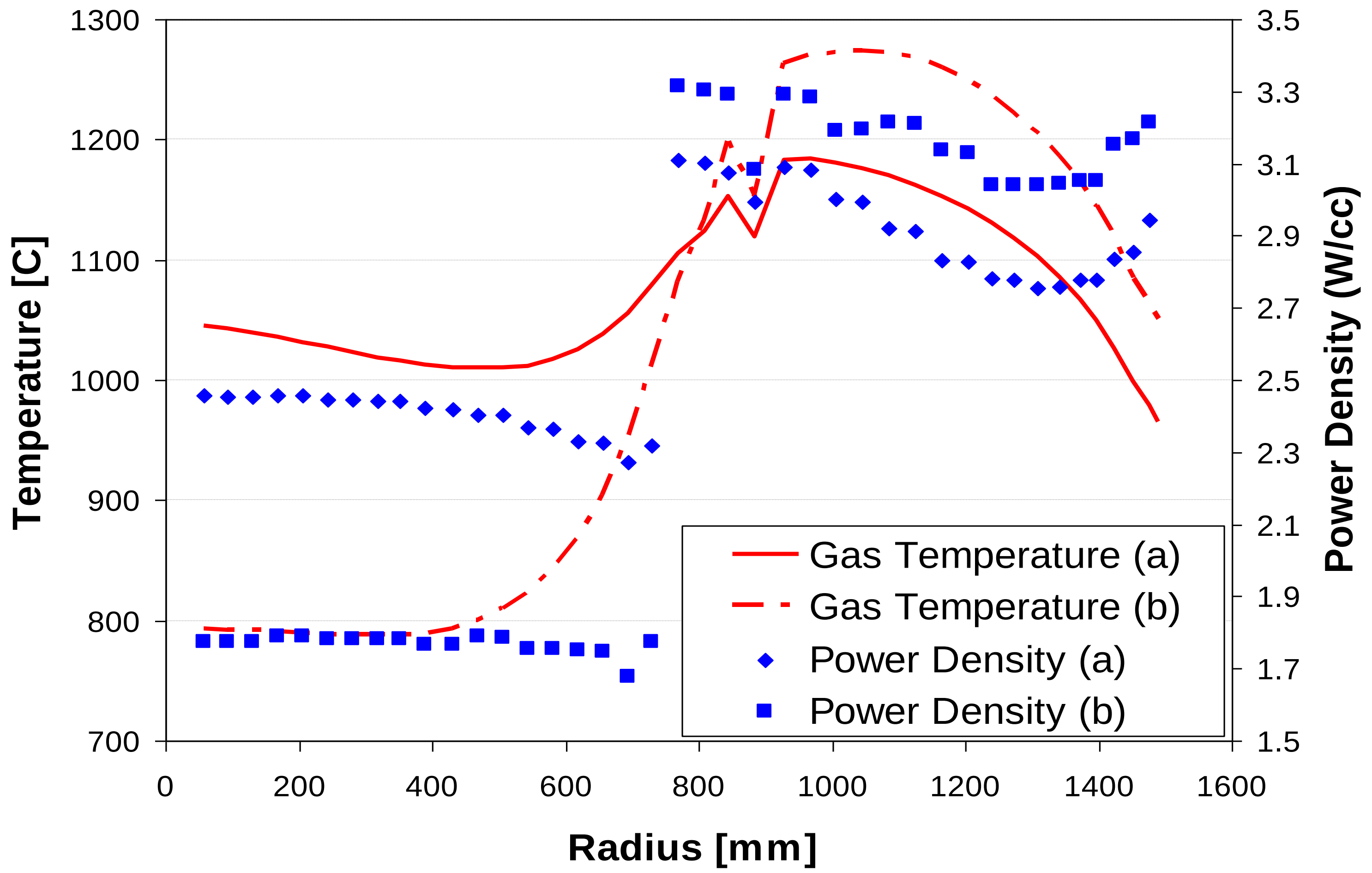
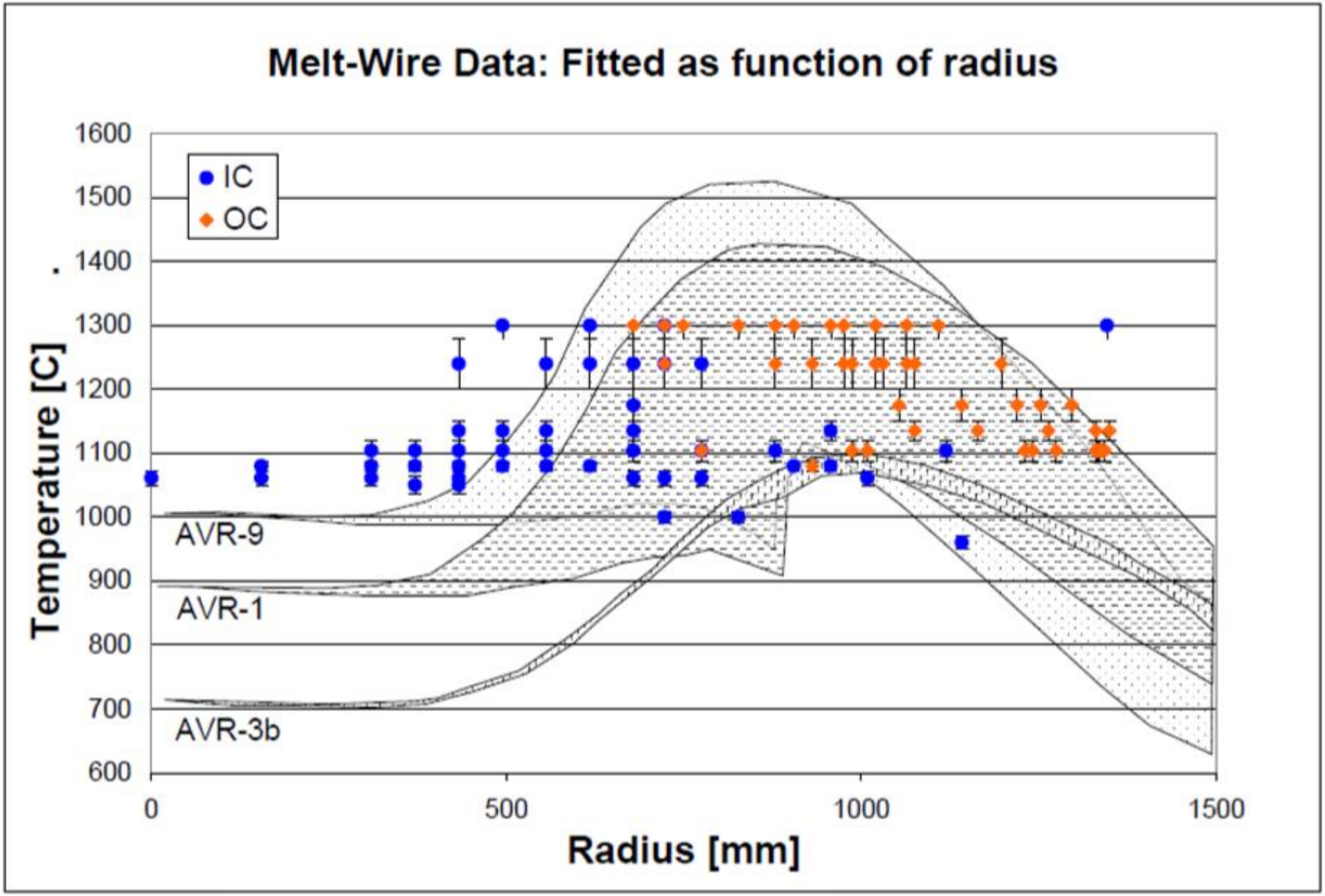


Conclusion

- Analysis shows bypass flows played a significant role in the AVR flow distribution
- Detail 3D thermo-hydraulic analysis is required for accurate predictions of flows and temperatures
- Uncertainties in AVR complicates comparison
  - ‘As-built’ geometry
  - Interpretation of measurements
- NEW: Meltwire measurements planned for HTR-10



Proceedings of HTR 2010 Prague, Czech Republic, October 18-20, 2010 Paper 193

The re-evaluation of the AVR melt-wire experiment with specific focus on different modelling strategies and simplifications. CF Viljoen , RS Sen

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Pebble Compaction…. Earth Quakes

Pebble Beds and earthquakes

- The impact of earthquakes on the PBMR design is investigated as part of the safety case
- Shaker-table experiments (SAMSON)
  - located at the HRG (Hochtemperatur-Reaktorbau GmbH) site at Jülich, Germany
  - used to postulate conservative compaction densities and times for use in the safety studies
- Focus of this study:
  - compaction of the pebble-bed or fuel region only
  - no radial disturbance in the core cavity dimensions - excluded by the core structure and graphite reflector design
  - change in the bulk or average packing density during an earthquake
  - Study core-neutronics and thermal-hydraulics behaviour of a postulated SSE

SAMSON Facility

SAMSON experiments at 0.4 g

- 0.61 -> 0.613 (5 seconds)
- 0.61 -> 0.616 (15 seconds)

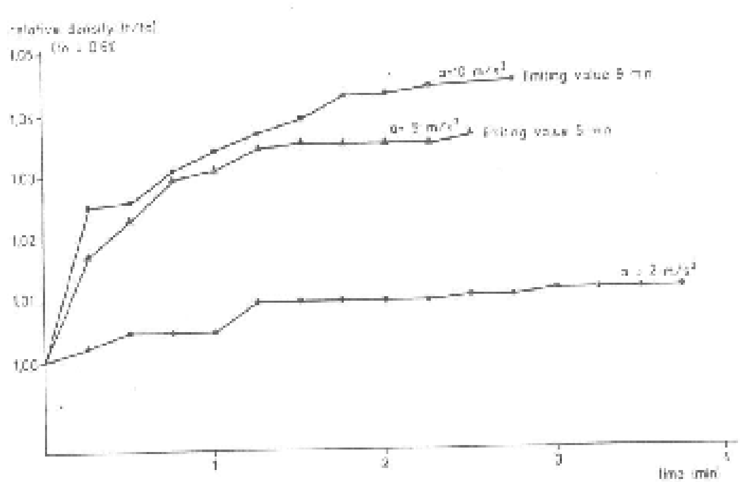


Abb. 4: Increase of the packing fraction f in a pebble bed by vibrating 1,5 m silo: 30 mm graphite balls

PBMR400 SSE postulated event

Postulate:

- Only effect is pebble bed compaction
- Decrease in pebble bed or core effective height
- Very conservative assumptions for concept design
  - Packing fraction increases: i) 0.61 -> 0.62 ii) 0.61 -> 0.64
  - No control rod movement
- Compaction duration: i) 5 seconds ii) 15 seconds
  - typical range for the duration of strong shaking that results from large earthquakes
- Includes a PLOFC and DLOFC (beyond design base)

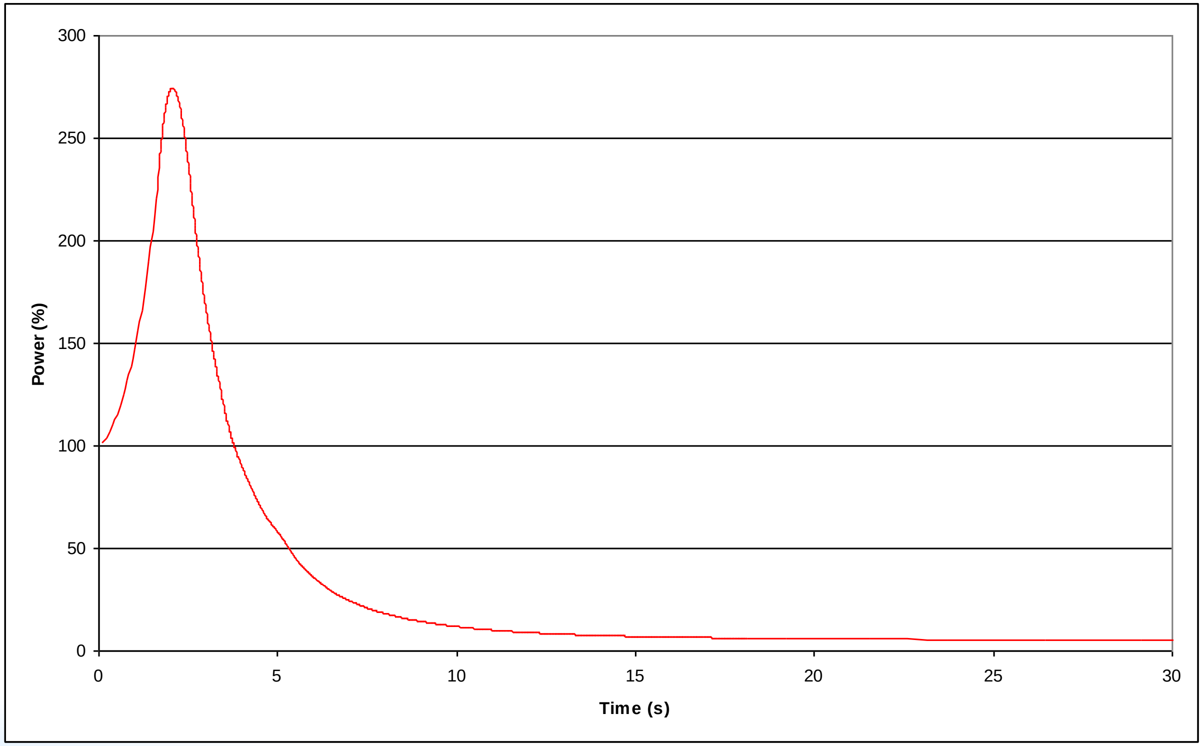
- Reactivity increase due to:
  - Denser packing of fuel spheres
  - Reduction of control rod effectiveness

Phenomena and restrictions

- The two major phenomena:
  - neutronic response of the fuel due to the bed compaction (streaming, leakage, spectrum changes, temperature feedback)
  - changes in the heat transfer (pebble bed packing fraction, reduced core height)
- quantify the changes in:
  - the core reactivity
  - fission power
  - material temperatures
  - fuel heat-up rate during the power excursion

Actual SSE results:

- SSE – Fission Power as a % of the Steady State Values (0 s to 30 s) with RPS Trip Initiated on the Reactor Power
- Control rod insertion begins at 1.73 s as a result of the power SCRAM set point



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Thank you!