Benchmarking FRAT: Chinese code for TRISO fuel performance analysis

NATIONAL CENTRE FOR NUCLEAR RESEARCH ŚWIERK

Nairi Baghdasaryan

Division of Nuclear Energy and Environmental Studies

nairi.baghdasaryan@ncbj.gov.pl

Supervisor: Tomasz Kozlowski

Fundusze Europejskie Wiedza Edukacja Rozwój

Rzeczpospolita Polska

Unia Europejska Europejski Fundusz Społeczny

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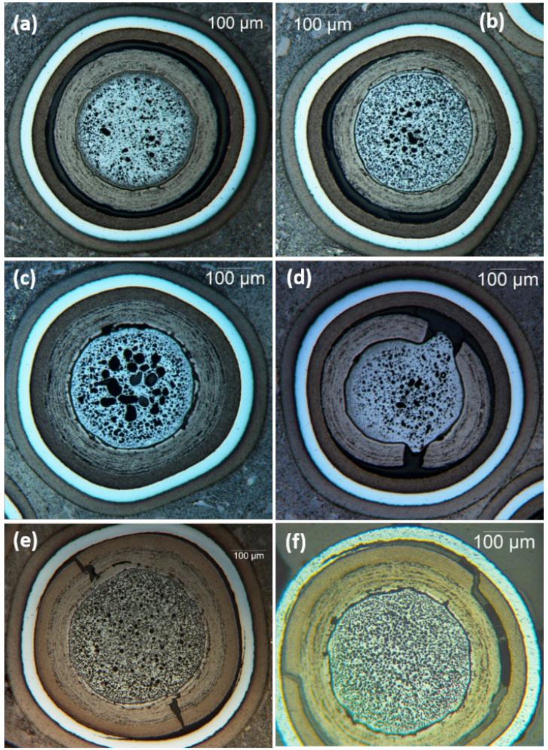
Introduction-1

What is covered in fuel performance analysis:

- Pressure build-up in buffer layer
- Heat generation and transfer from the kernel
- Anisotropy effects
- Kernel migration
- Fission product build-up and release
- Chemical effects
- Thermomechanical behavior (irradiation induced)
- Fuel failure mechanisms and failure fraction

Questions need an answer:

- What is the allowable max burnup level ?
- What are the optimal geometric and material properties ?
- How operational conditions affect of fuel performance ?
- How much radioactive material will go out from TRISO during operational/accidental conditions ?
- What is the failure fraction of the fuel particles?
- Etc …



Examples of various AGR-1 irradiated particle microstructures [2]

Introduction-2

Why quantifying uncertainties is important?

- Manufacturing source of uncertainties (geometry and materials)
- Model based uncertainties
- Many phenomena are cross-dependent
- Small uncertainty in one model can have a big impact on the accuracy of another model
- Is current design of TRISO the most optimal?
- Doing experiments are not cheap and not fast
- Even experiments are uncertain

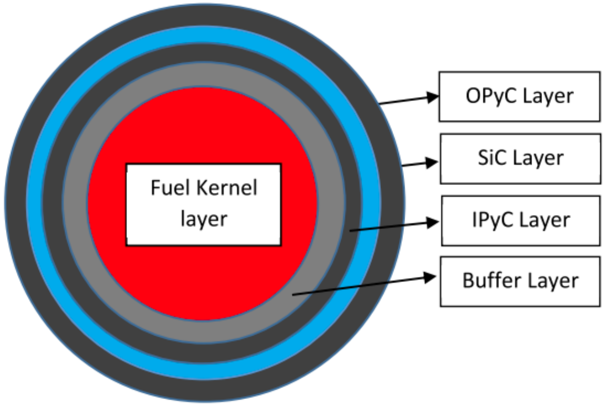


Fig. 1. The geometry of the TRISO fuel particle.

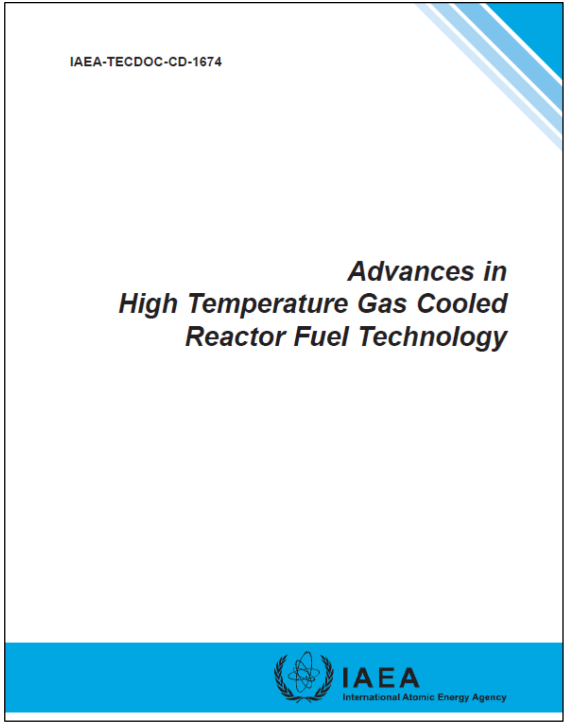
CRP Benchmark-1

A key part of CRP-6 includes benchmark calculations with fuel performance models under normal HTGR operating conditions. The normal operation benchmarking has been structured in two phases:

- Phase 1: a series of simplified analytical benchmarking problems have been established as a way to 'calibrate' the codes and/or models.
- Phase 2: the codes and/or models will be used to calculate more complicated benchmarks of actual experiments that have been completed and of planned experiments.

Total benchmarking problems: 13

Participating countries: USA (INL,GA), France,
UK, Russia, Korea, Turkey


IAEA-TECDOC-CD-1674

Advances in
High Temperature Gas Cooled
Reactor Fuel Technology

IAEA
International Atomic Energy Agency

Advances in HTGR Reactor Fuel Technology," IAEA-TECDOC-CD-1674 [3]

Selected Cases For Benchmarking

Analytical Problems

Parameter	Case 5: TRISO 350 μm kernel	Case 6: TRISO 500 μm kernel	Case 7: TRISO High BAF
Oxygen to Uranium ratio	2	2	2
Carbon to Uranium ratio	0	0	0
U-235 enrichment (wt%)	10	10	10
Kernel diameter (μm)	350	500	500
Buffer thickness (μm)	100	100	100
IPyC thickness (μm)	40	40	40
SiC thickness (μm)	35	35	35
OPyC thickness (μm)	40	40	40
Kernel density (Mg/m^3)	10.8	10.8	10.8
Buffer density (Mg/m^3)	0.95	0.95	0.95
IPyC density (Mg/m^3)	1.9	1.9	1.9
SiC density (Mg/m^3)	3.20	3.20	3.20
OPyC density (Mg/m^3)	1.9	1.9	1.9
IPyC BAF	1.03	1.03	1.06
OPyC BAF	1.03	1.03	1.06
Irradiation duration (efpd)	1000	1000	1000
End of life burnup (% FIMA)	10	10	10
End of life fluence (10^{25} n/m^2, E > 0.18 MeV)	3	3	3
Constant irradiation temperature (K)	1273	1273	1273
Ambient pressure (MPa)	0.1	0.1	0.1
Results to be compared	Inner surface SiC and IPyC tangential stress historie		

Experiment-Based Problems

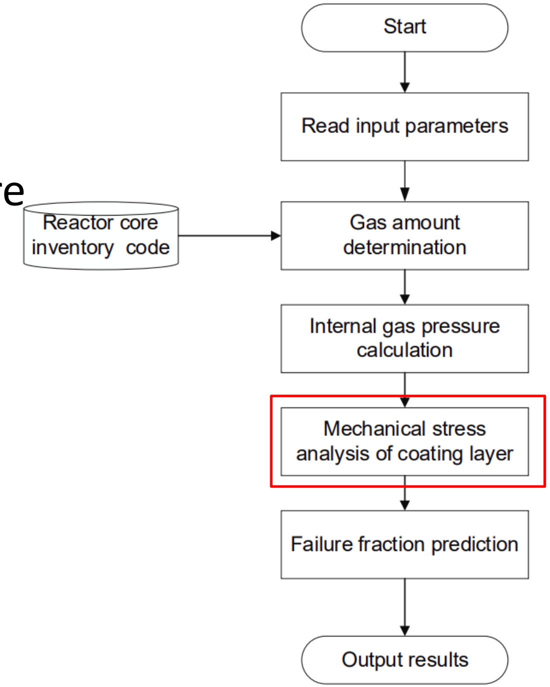
Parameter	Case 9: HRB-22	Case 10: HFR-K3/B/2	Case 11: HFR-P4/3
Oxygen to Uranium ratio	2	2	2
Carbon to Uranium ratio	0	0	0
U-235 enrichment (wt%)	4.07	9.82	9.82
Kernel diameter (μm)	544 ± 9	497 ± 14	497 ± 14
Buffer thickness (μm)	97 ± 13	94 ± 10	94 ± 10
IPyC thickness (μm)	33 ± 3	41 ± 4	41 ± 4
SiC thickness (μm)	34 ± 2	36 ± 2	36 ± 2
OPyC thickness (μm)	39 ± 3	40 ± 2	40 ± 2
Kernel density (Mg/m^3)	10.84	10.81	10.81
Buffer density (Mg/m^3)	1.10	1.00	1.00
IPyC density (Mg/m^3)	1.85	1.88	1.88
SiC density (Mg/m^3)	3.20	3.20	3.20
OPyC density (Mg/m^3)	1.85	1.88	1.88
IPyC BAF	1.00	1.053	1.053
OPyC BAF	1.00	1.019	1.019
Irradiation duration (efpd)	89	359	351
End of life burnup (% FIMA)	4.79	10	14
End of life fluence (10^{25} n/m^2, E > 0.18 MeV)	2.1	5.3	7.2
Time-average, volume-average irradiation Temperature (°C)	1303	1073	1335
Ambient pressure (MPa)	0.1	0.1	0.1
Comparison metric	end of life failure fraction		

About FRAT

Fissions product Release Analysis Tool

The methods, models, and libraries used by FRAT[4] are outlined below:

- The amounts of fission gases provided by the reactor core inventory code are adopted by FRAT.
- FRAT uses ideal gas law or Redlich-Kwong gas state equation for internal gas pressure calculation.
- FRAT adopts Miller's analytical solution for stress-strain-displacement analysis for TRISO-coated fuel particle.
- Weibull strength theory is adopted by FRAT for particle failure fraction determination.



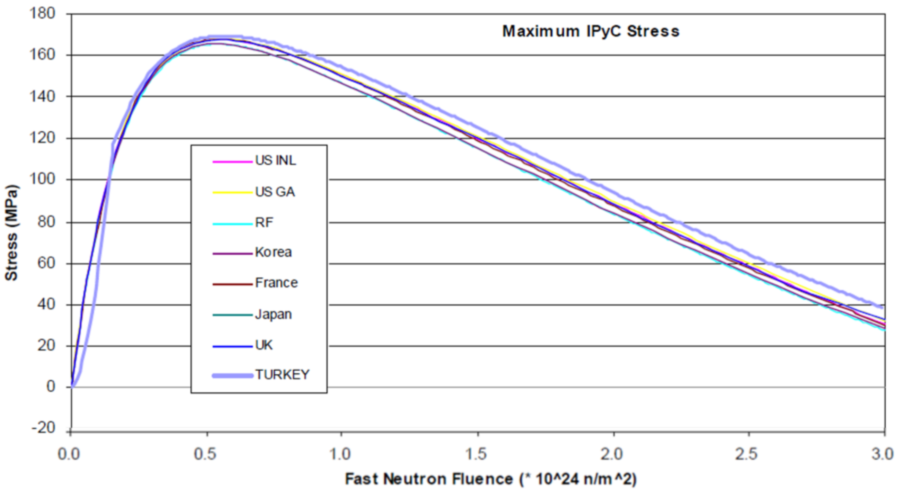
Institute of Nuclear and New Energy Technology, Tsinghua University

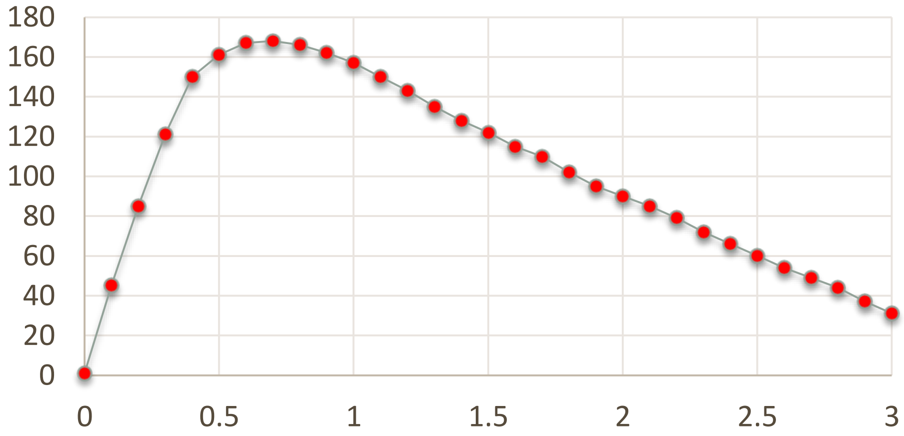
Results: Case 6 - Part1

Case description:

A full three layer (TRISO) coated particle with a 500µm diameter kernel under realistic service conditions:

- IPyC/SiC composite with constant creep and neutron dose dependent swelling
- Linearly raising pressure dependent on burnup (from 0 to 26.20 MPa).



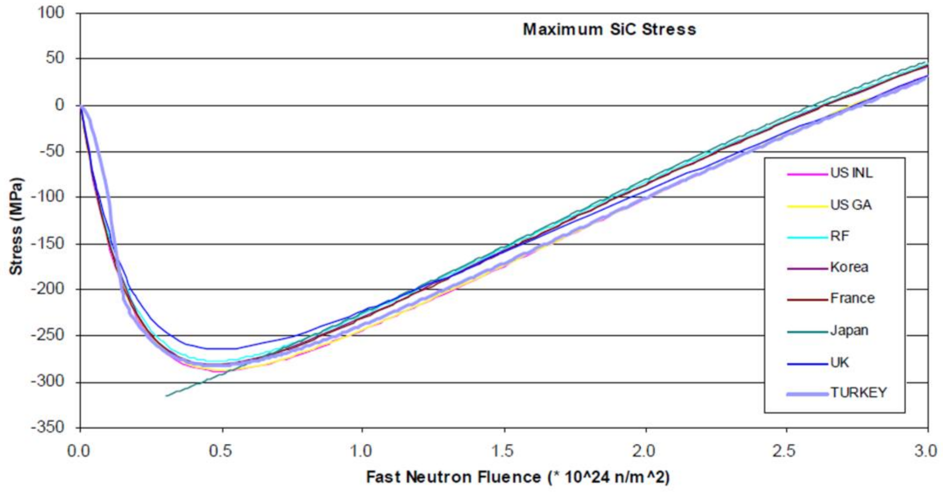


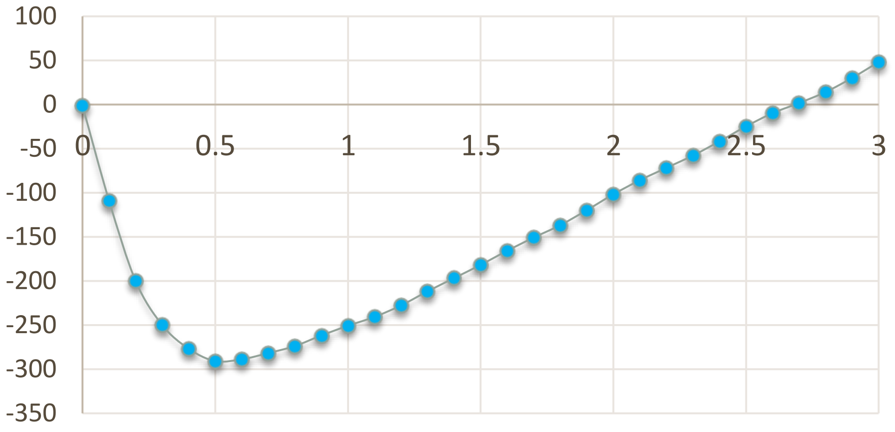
Results: Case 6 - Part 2

Case description:

A full three layer (TRISO) coated particle with a 500µm diameter kernel under realistic service conditions:

- IPyC/SiC composite with constant creep and neutron dose dependent swelling
- Linearly raising pressure dependent on burnup (from 0 to 26.20 MPa).





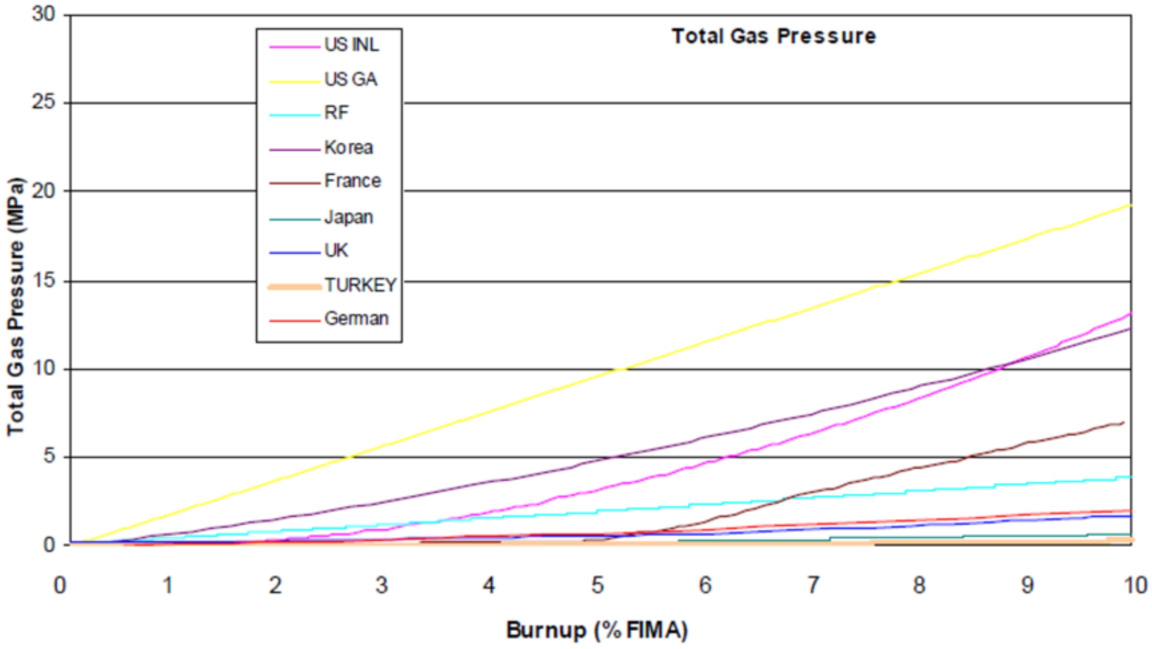
Results: Case 2

Case description:

HFR-K3/2 experiment [5]

LEU TRISO coated particles with a mean kernel diameter equal to 497 µm in a pebble irradiated 359 days until 10% FIMA and $5.3 * 10^{25} \text{n}/m^2$

HFR-K3/2 : Total gas pressure as a function of burnup



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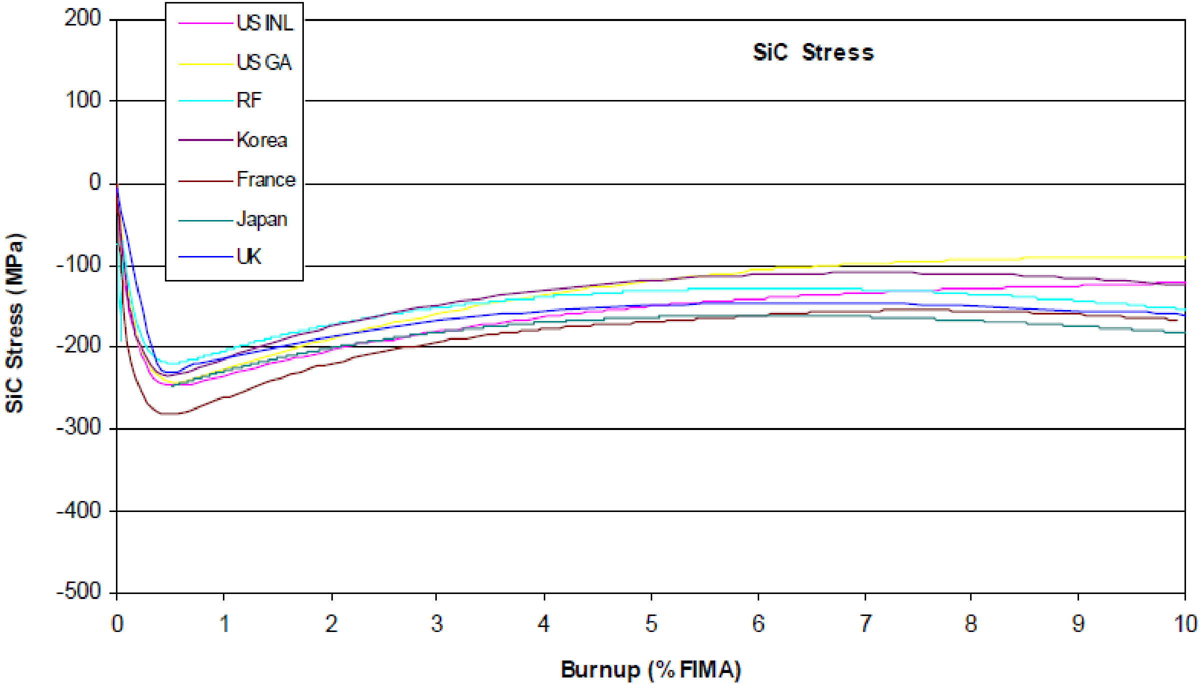
HFR-K3/2 : SiC stress as a function of burnup



Summary

To conclude:

- Fuel performance analysis of TRISO fuel are important for safety and licensing perspectives
- Uncertainty quantification in fuel performance analysis is complex but very important
- Preliminary results show that FRAT code is suitable for such analysis, but further verification is needed

Literature

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2. S.A. Ploger, P.A. Demkowicz, J.D. Hunn, J.S. Kehn, Microscopic analysis of irradiated AGR-1 coated particle fuel compacts, Nuclear Engineering and Design, 271, 221-230 (2014)
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Thank you for attention

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