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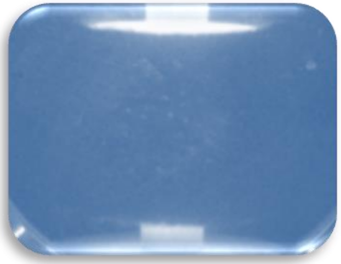
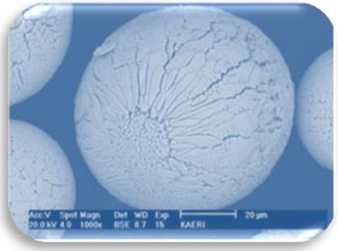
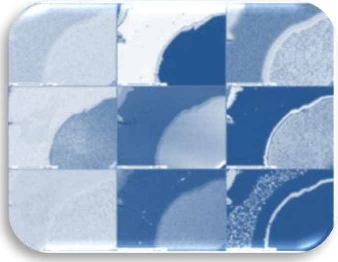
Transactions of the Korean Nuclear Society Autumn Meeting
Changwon, Korea, October20-22, 2021







Development of Low-Temperature Fuel Performance Analysis Code for Micro Ultra Long life Lead-cooled Fast Reactor

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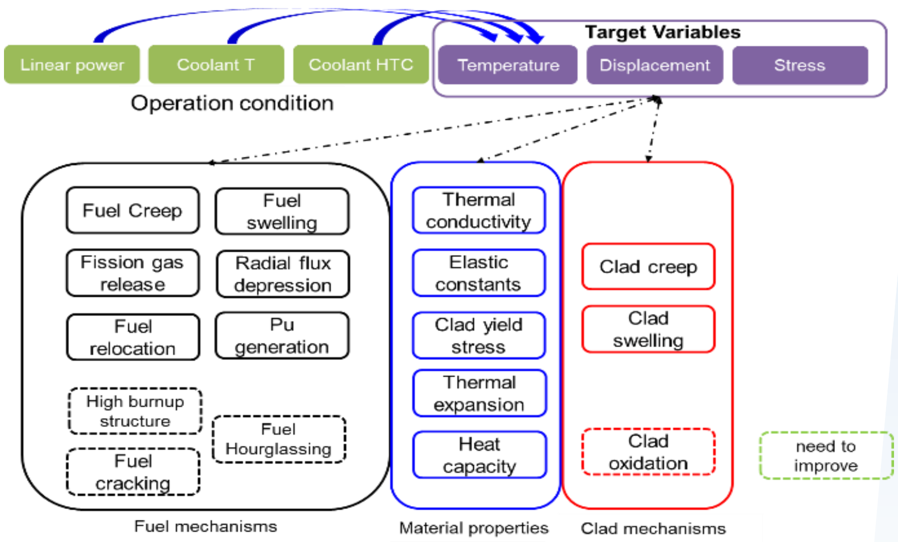
Contents

Introduction

<Objectives of the study>

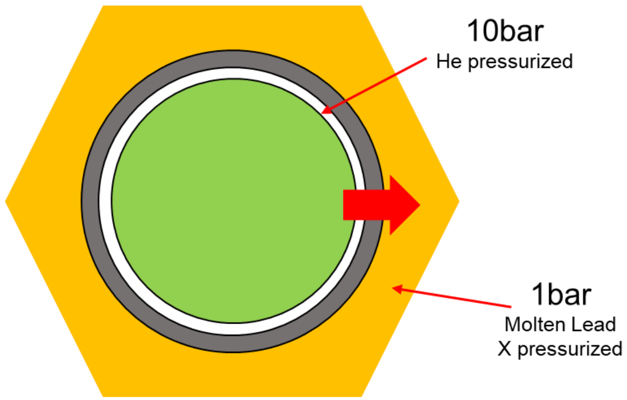
Development of **Low Temperature Fuel (<1200K)** performance evaluation code of the long life Micro **LFR**

- Fuel performance code for LFR FRAPCON-KAIST-1.0 has been developed based on FRAPCON-4.0
- The evaluation of the thermal and mechanical performance of nuclear fuel during steady-state operation for 30 years was conducted

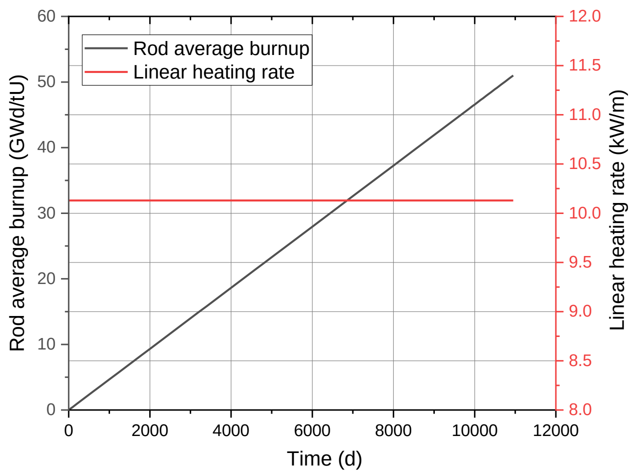


Overall code structures and target variables

Simulation Condition



Schematic cross section image of the fuel rod



Reactor burnup and power profile

Table 1. Fuel rod and LFR core design in this study

Design Factor	Design Value
Fuel Enrichment (%)	9.75
Fuel rod outer diameter / Cladding thickness(mm)	20.0 / 0.95
Cladding material	15-15Ti
Fill gas material	He
Initial Fill gas pressure (bar)	10
Plenum length (cm)	10
Fuel rod length (cm)	155
Core thermal power(MWt)	60
Average linear heat rate(kW/m)	10.43
Coolant properties	
Coolant Pb/Bi composition (wt%)	44.5/55.5
Coolant inlet/outlet temperature(℃)	250.0/350.0
Mass flux of coolant (kg/m²·s)	5534.76

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Important Modified Material Properties

Modified Material properties

- Material properties modification in FRAPCON-4.0 Light Water Reactor (LWR) normal operation fuel performance code

Module	Modified Material	Modified Properties
Fuel	UO_2	Swelling, Pu generation
Cladding	Zircaloy → 15-15 Ti	Swelling Thermal conductivity, Heat capacity, Thermal expansion, Transition temperature Modulus, Creep (Thermal, Irradiation)
Coolant	Water → LBE	Constant heat transfer coefficient Heat capacity, mass flux

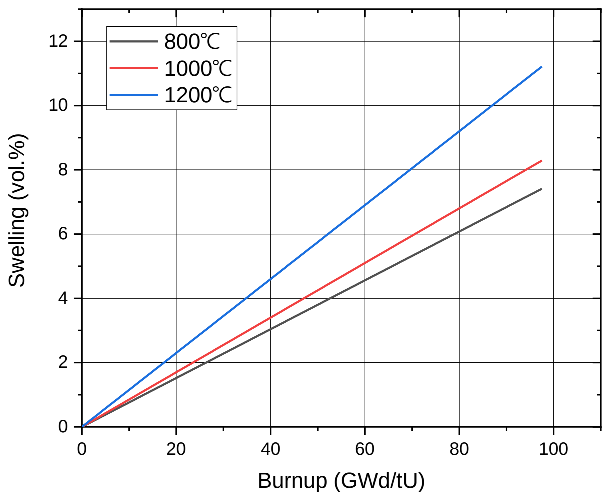
Fuel material property	**Low Temperature Applicability**	**Fuel mechanism**	**Low Temperature Applicability**
Thermal conductivity	O	Swelling	X
Thermal expansion	O	Fission gas release	O
Melting temp.	O	Densification	O
Specific heat	O	Radial burnup depression	X (Fast spectrum applicability)
Enthalpy	O		
Emissivity	O		

Fuel Swelling

- When the temperature of the nuclear fuel is low, the fission gas does not sufficiently grow at the grain boundary and is destroyed by the fission fragments that contain high kinetic energy

- Therefore, in the case of fuel operated at a low temperature below 1200℃ and high burnup, fission gas is hardly released and exists in the form of a supersaturated solid solution in the nuclear fuel.

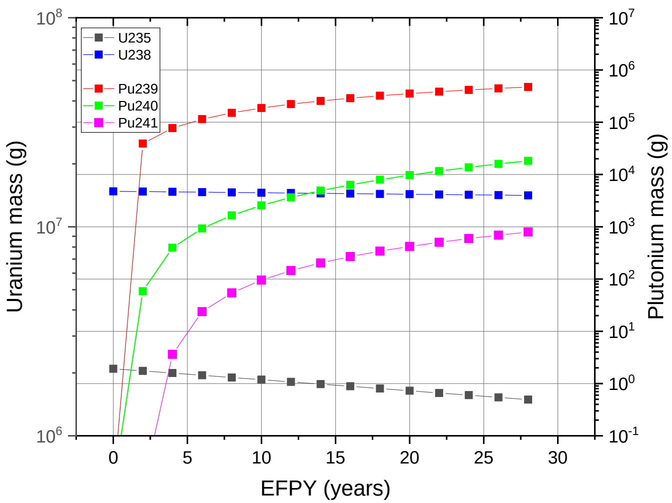
Module	Previous model in FRAPCON -4.0	Low temperature model
Volumetric Swelling per 1 GWd/tU	Solid swell (0.062 vol.%) + Gas swell (0 vol.% @T<1000℃)	0.076vol.% @ 800℃ 0.085vol% @1000℃ 0.115vol.% @ 1200℃
Reactor type	LWR	LWR
Reference	[1]	[2]

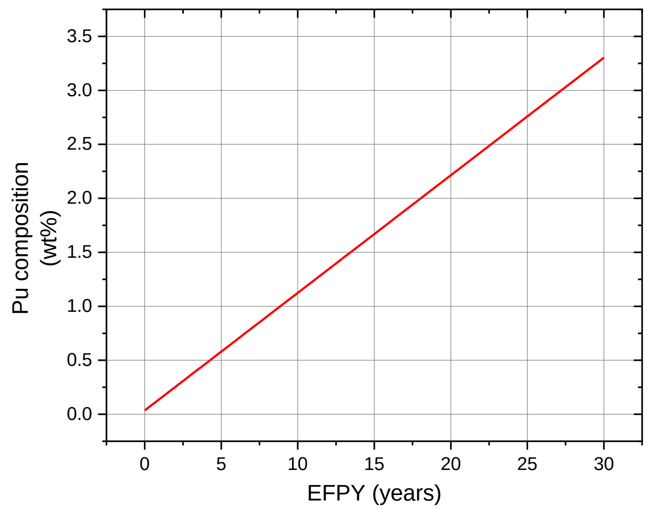


[1] Geelhood, K. J., et al. "A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup." (2015).
[2] Dollins, C. C., and H. Ocken. "A Fission Gas Swelling Model Incorporating Re-Solution Effects." *Nuclear Applications and Technology* 9.2 (1970): 141-147.

Pu generation

Pu generation data were obtained by core neutronics calculation [3]



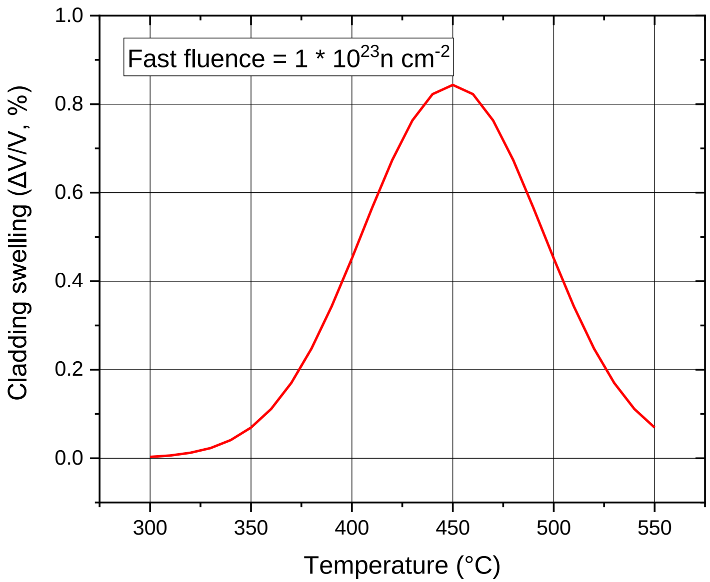


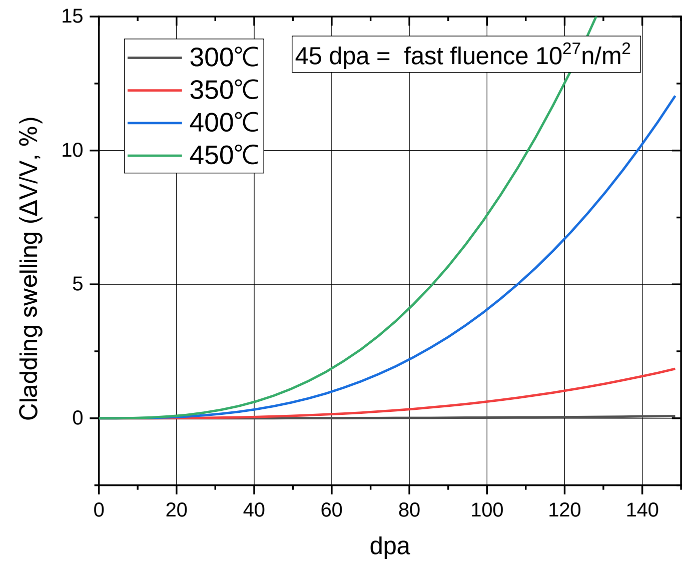
- **Fuel property change** as Pu composition change were implemented (density, heat capacity, thermal expansion, enthalpy, modulus, thermal conductivity)

- **No fission gas release enhancement effects** are reported for MOX fuel operated **at low power and low temperature**, resulting in low (<3 %) fission gas release fractions [4,5]

[3] Nguyen, Tung Dong Cao, et al. "MicroURANUS: Core design for long-cycle lead-bismuth-cooled fast reactor for marine applications." International Journal of Energy Research (2021).
[4] Walker, C. T., W. Goll, and T. Matsumura. "Effect of inhomogeneity on the level of fission gas and caesium release from OCOM MOX fuel during irradiation." Journal of Nuclear Materials 228.1 (1996): 8-17.
[5] White, R. J., et al. "Measurement and analysis of fission gas release from BNFL's SBR MOX fuel." Journal of nuclear materials 288.1 (2001): 43-56.

Cladding Swelling





$$\frac{\Delta V}{V}\,[\%] = 1.5\,\times 10^{-3}\exp\left[-2.5\left(\frac{T[℃]-450}{100}\right)^{2}\right]\phi^{2.75}$$

$\phi\left[\frac{10^{22}\,n}{cm^2}\right]$: fast fluence, $T[℃]$: cladding temperature

- Highest swelling values occur at 450 ℃
- Temperature dependence of the clad volume variation (△V/V) is in a Gaussian form
- Peak cladding damage is 120 dpa at EOL

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[6] M.B. Toloczko, F.A. Garner, C.R. Eiholzer, Determination of the creep compliance and creep-swelling coupling coefficient for neutron irradiated titanium-modified stainless steels at ~ 400°C, J. Nucl. Mater. (1992).

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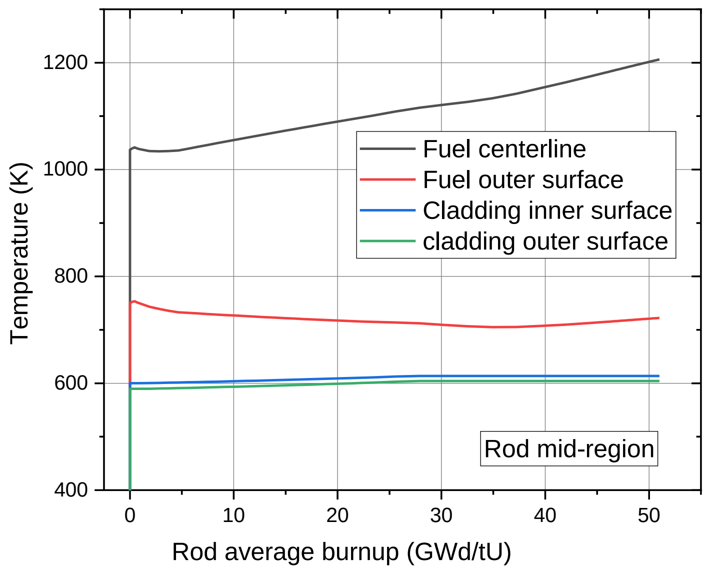
Fuel Performance Evaluation Results and Validation

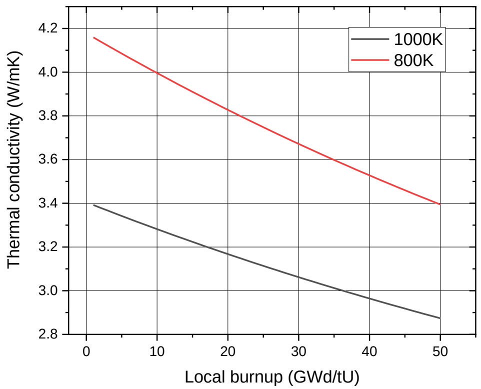
Fuel temperature



Thermal conductivity

$$K = \frac{1}{A(x) + B(x)T + f(Bu) + g(Bu)h(T)} - \frac{C}{T^2}\exp(-\frac{D}{T})$$

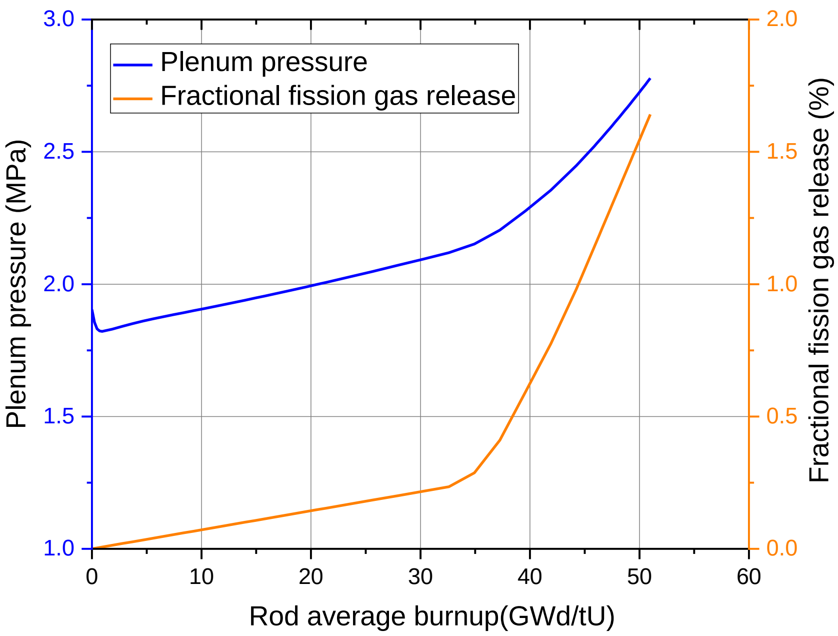
x = deviation from stoichiometry = 2 - O/M
A(x) = 2.85 + 0.035
B (x) = (2.86-7.15x)*10^{-4}
f(Bu) = effect of fission products (solution)
= 0.00187 *Bu
g(Bu) = effect of irradiation defects
= 0.038*Bu
h(T) =temperature dependence
of annealing on irradiation defects
= $(1+396\exp(-Q/T))^{-1}$



- Fuel temperature increases due to “thermal feedback”

1. Irradiation induced fuel thermal conductivity degradation
2. Gap conductance decrease by fission gas release

Fission Gas Release



- **Low-Temperature Fission Gas Release (FGR) Empirical Model**

$$F_{LT} = 7 \times 10^{-5}\, BU + C$$

F_{LT} = fission gas release fraction by low temp. model
BU = local burnup in GWd/tU
C = 0; for BU ≤ 40 GWd/tU
= 0.01(BU-40)/10; for burnup > 40 GWd/tU and F_{LT} ≤ 0.05
(enhanced release by formation of restructured grain)

➢ FGR showed linear behavior in both sections based on about 35GWd/tU
(it is the rod average burnup, not local burnup)

➢ Stiff increase after 35 GWd/tU due to the C value change
➢ Maximum FGR: 1.7%

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[7] J. Turnbull. Presentation at the IAEA Research Co-ordination Meeting on Fuel Modelling at Extended Burnup (FUMEX) , Bombay, India, 1-5 April, 1996.

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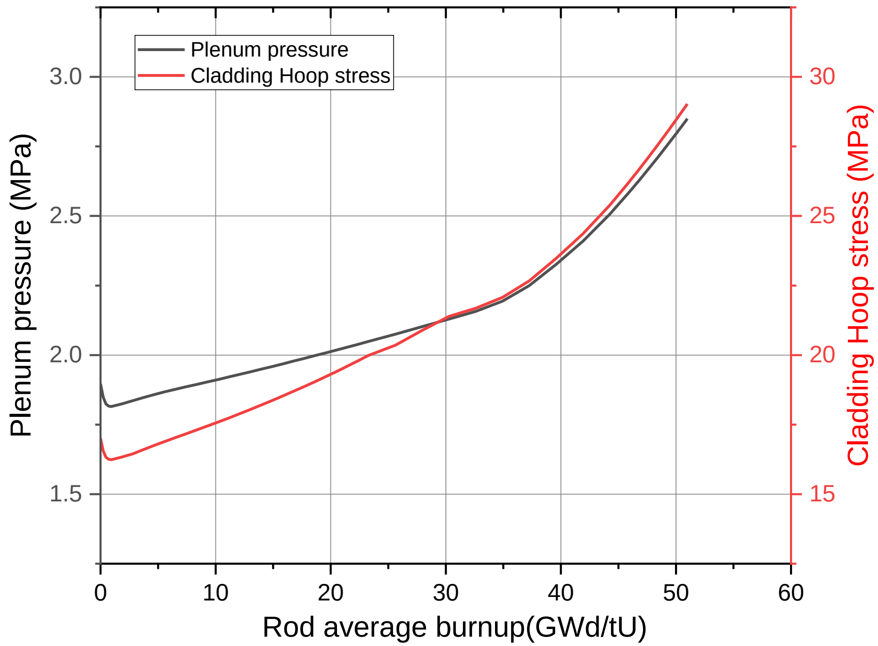
Mechanical analysis



- Low hoop stress level (< 30MPa) that insure cladding integrity due to large cladding thickness and low fission gas release
- Cladding temperature is low (~350℃) that ensures extremely low cumulative damage fraction CDF

($3.65*10^{-18}$,CDF < 10^{-5} – general criterion)

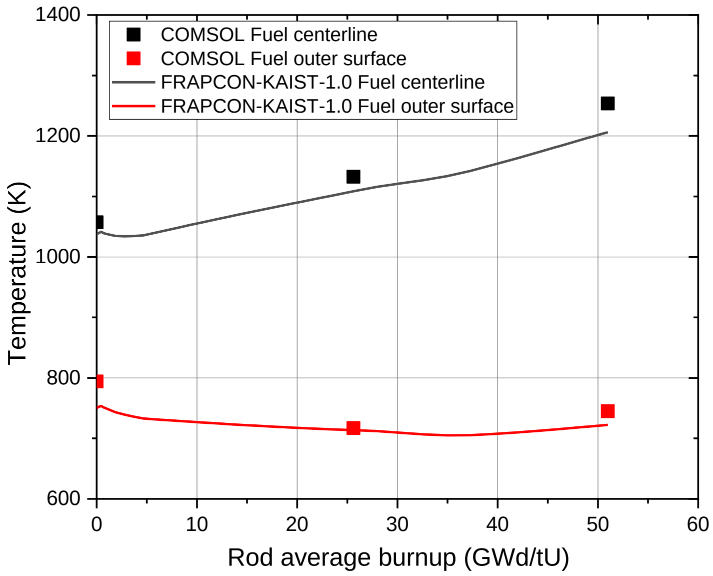
$$LMP = T[16.0 + log_{10}(t_R)] = (2060 - \sigma_H)/0.095$$

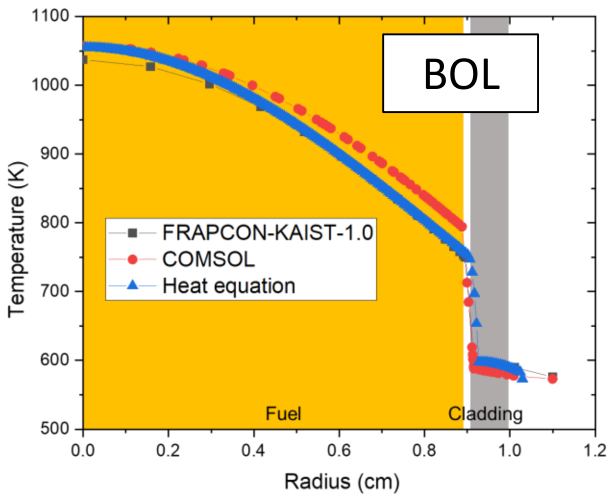
t_R = rupture time (days),σ_H = hoop stress (MPa), T = cladding temperature

$$CDF = \int_0^t \frac{dt}{t_R}$$

[8] Luzzi, Lelio, et al. "Application of the TRANSURANUS code for the fuel pin design process of the ALFRED reactor." Nuclear Engineering and Design 277 (2014): 173-187.

Thermal Validation





- Thermal analysis results were cross-checked by COMSOL and heat equation

➢ **Code results** show less than 10K calculation difference with **Heat equation**

➢ Relatively larger differences between COMSOL arise from the difference in number of radial nodes and meshes used in each calculation

Conclusions

Conclusions

- LWR fuel performance analysis code FRAPCON-4.0 was modified to be applied to micro LFR

- Material properties of the the **low temperature UO_2, austenitic stainless cladding, and LBE coolant** fast reactor were changed.

- It has been shown that fuel can be **maintained at temperatures as low as 1225 K** or less **for 30 years** of full power operation.

- Maximum fractional fission gas release does not exceed 1.7% which can reduce rod internal pressure build-up and allow high initial He pressurization


Thank you