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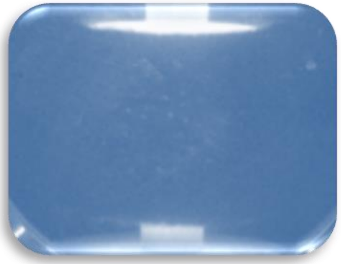
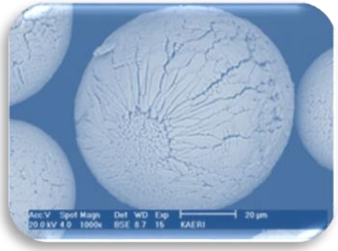
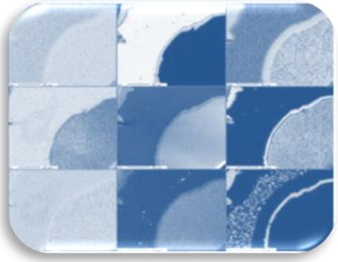
Korean Nuclear Society
Autumn Meeting
Changwon, Korea
Dec 16-19, 2020







Low Linear Power Oxide Fuel Performance Evaluation for Micro Lead-cooled Fast Reactor

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Contents

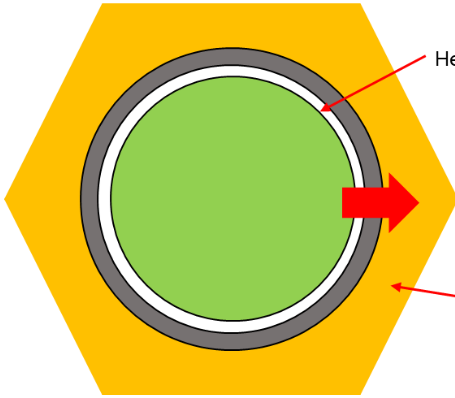
Introduction

<Objectives of the study>

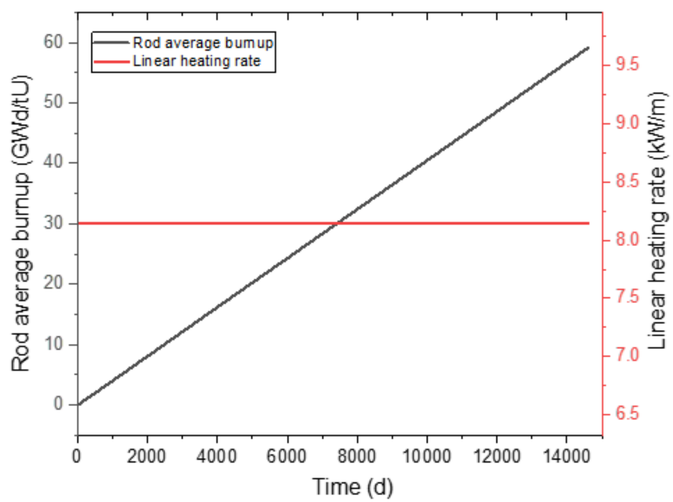
Development of **Low Temperature Fuel (~800K)** performance evaluation code of the long life Micro **LFR**

- **The lead-cooled fast reactor (LFR) is considered as one of the most promising new generation fast nuclear reactors.**
- **Three reference systems were adopted by LFR-provisional System Steering Committee (pSSC) that include ELFR, ALFRED (EU), BREST-OD-300 (Russia) and SSTAR (USA) and they are utilizing MOX or mixed nitride fuel.**
- **The preliminary design of ultra-long life micro LFR is currently being studied in Korea as a nuclear propulsion system for icebreakers that has a linear heat generation rate as low as 1/10 scale than conventional fast reactor.**
- **At low temperatures under 1200K, fuel irradiation properties change greatly.**
- **Therefore, development of a new fuel performance analysis code for low temperature fuel in LFR is essential.**

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 material	UO_2
Fuel rod outer diameter / Cladding thickness(mm)	20.0 / 0.95
Cladding material	SS316 SS316Ti
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)	8.14
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

Important Modified Material Properties

Fission gas release

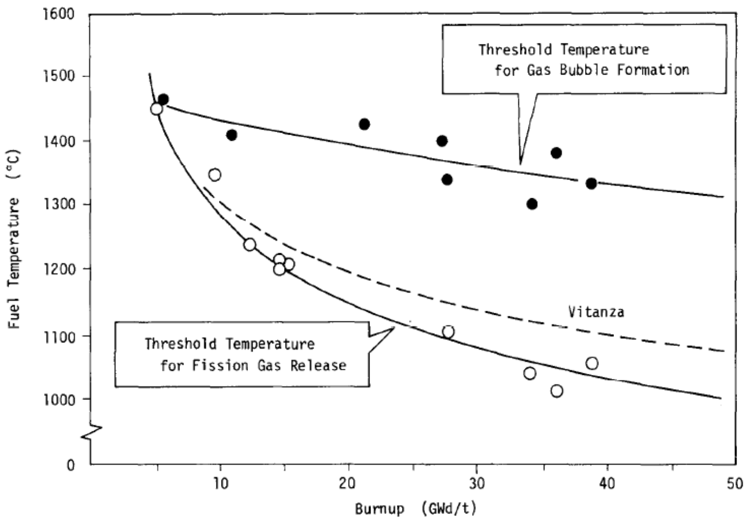
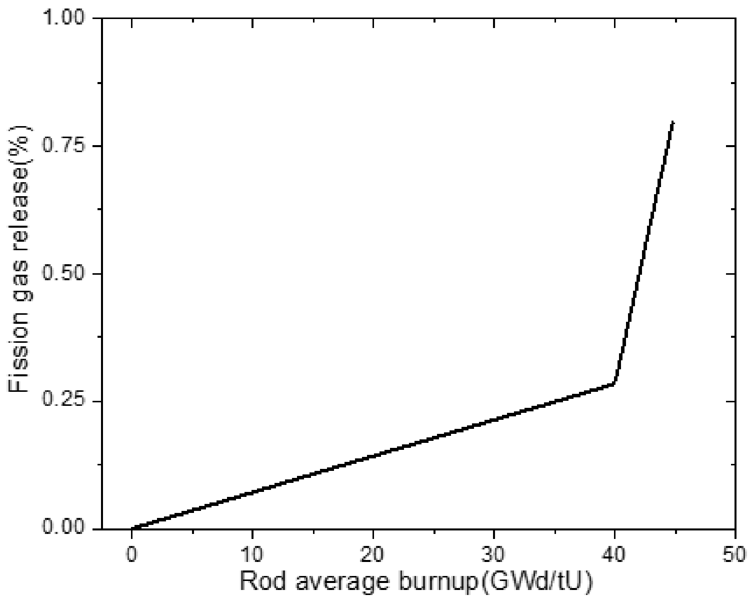


Fig. 5. Threshold temperature for fission gas release and bubble formation as a function of burnup. Broken curve: Threshold temperature for fission gas release by Vitanza et al. [21].

JOYO MK-I MOX fuel

< Calculation results>



- **Threshold temp. for fission gas release Ts = 1000 ℃ for 50 GWd/tU**
- **Even considering the high burnup structure of the rim part, fission gas release will hardly occur**
- 0.75% fission gas release were calculated by default low temperature fission gas release model ➔ reasonable

$$F = 7\times10^{-5}\,BU + C$$

F = fission gas release fraction
BU = local burnup in GWd/MTU
C = 0; for $BU \le 40$ GWd/MTU
= $0.01(BU\text{-}40)/10$; for burnup > 40 GWd/MTU and $F \le 0.05$

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1) Ukai, S., et al. *Journal of Nuclear Materials* 151.2 (1988): 209-218.

Fuel Swelling

- **Low fission gas release ➔ high gaseous swelling**

- **When the temperature of nuclear fuel is low, fission gas does not grow sufficiently at the grain boundary and is destroyed by fission fragments.**
- **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.**
- **This causes high swelling of nuclear fuel, and it has been calculated from the existing literature that a volume expansion of 0.1% per 1 GWd/tU occurs (2).**

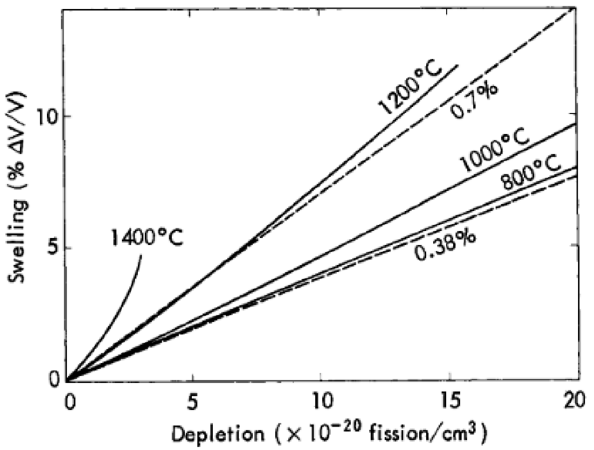
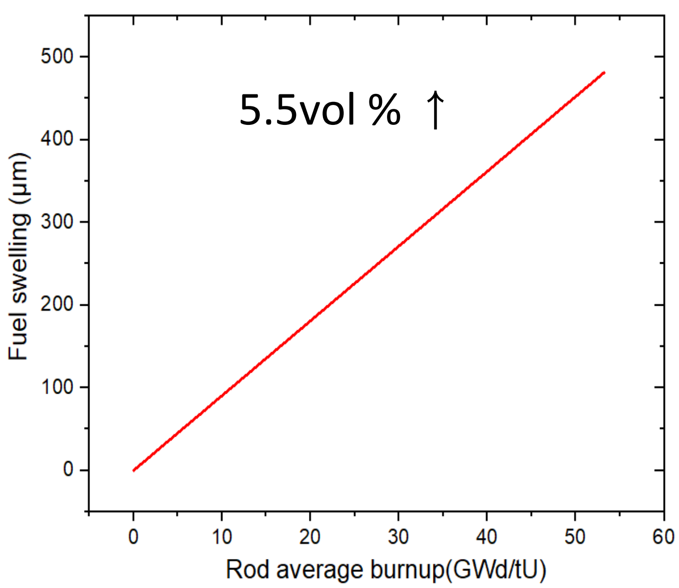


Fig. 1. The predicted swelling of UO_2 as a function of depletion and temperature based upon the re-solution swelling model. The limiting swelling rate in the absence of bubble growth (0.38%/10^{20} fission/cm^3) and the representative swelling rate (0.7%/10^{20} fission/cm^3) are also indicated.

(3)

< Implemented results>



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2) Guerin, Y. "Fuel performance of fast spectrum oxide fuel." (2012): 547-578.
3) C. C. Dollins ,Nuclear Applications and Technology (1970)

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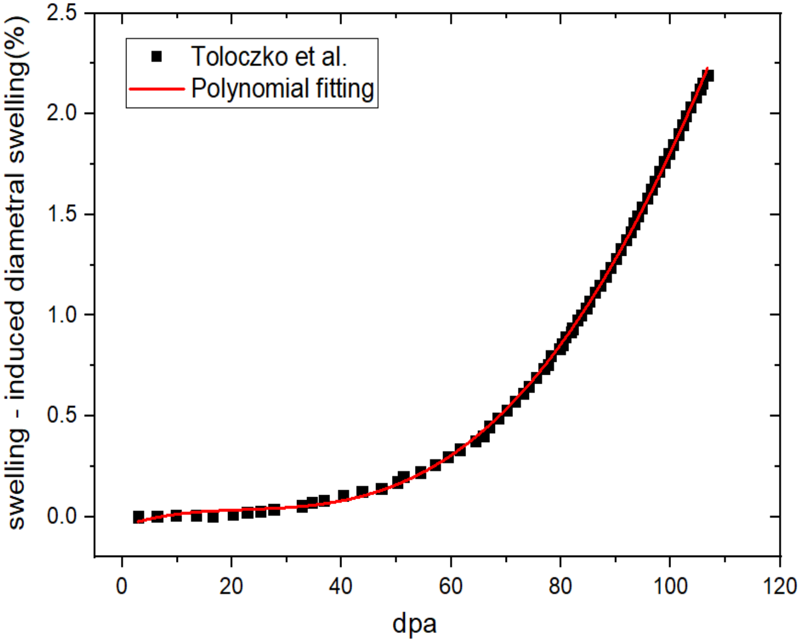
Cladding Swelling



$$A = -0.4899$$

$$B_1 = 0.0094,\ \ B_2 = -3.90818E-4,\ \ B_3 = 6.61546E-6,\ \ B_4 = -1.78659E-8$$

$$\frac{\Delta d}{d}[\%] = A + B_1(dpa) + B_2(dpa)^2 + B_3(dpa)^3 + B_4(dpa)^4$$

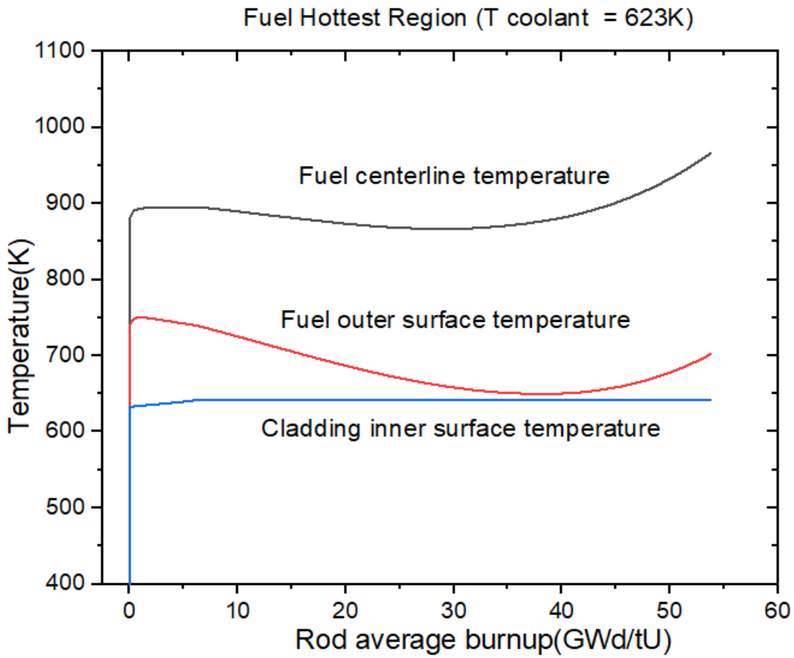
- **Data of swelling induced diametral strain of Ti modified type 316 steel at 400℃ were implemented by polynomial fitting in this code.**

- **The fitting function shows a good agreement with the original data.**

4) 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).

Fuel Performance Evaluation Results and Validation

Fuel temperature



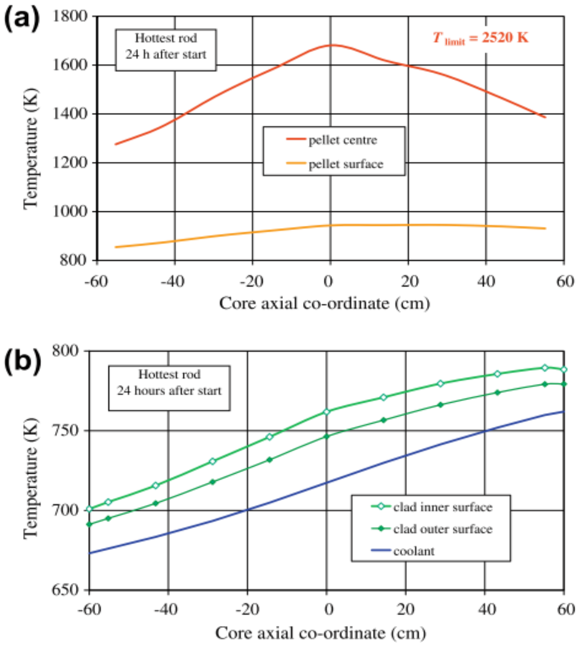


Fig. 3. Axial distribution of temperature in the fuel column (a) and in the cladding (b) of the hottest fuel rod at start. 5)

- Maximum 1000K centerline temperature
- Very low compared to about 1800K, the design operating temperature of the existing European ELSY project LFR

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5) Sobolev, Vitaly, E. Malambu, and H. Aït Abderrahim. "Design of a fuel element for a lead-cooled fast reactor." Journal of Nuclear Materials 385.2 (2009): 392-399.

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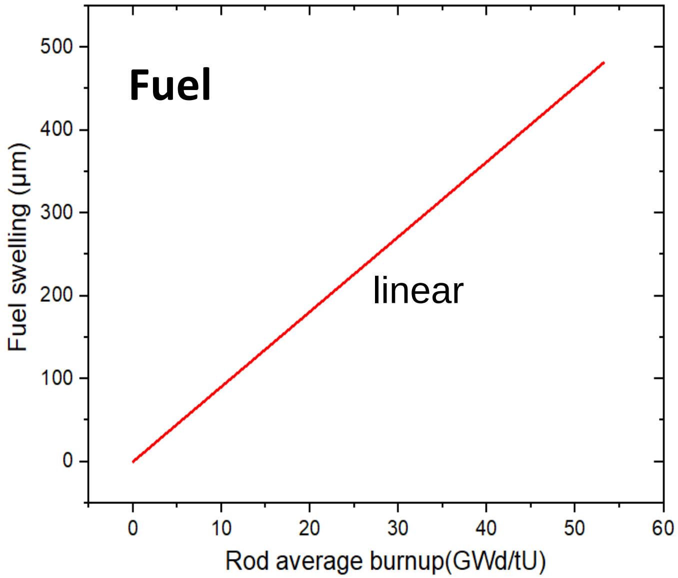
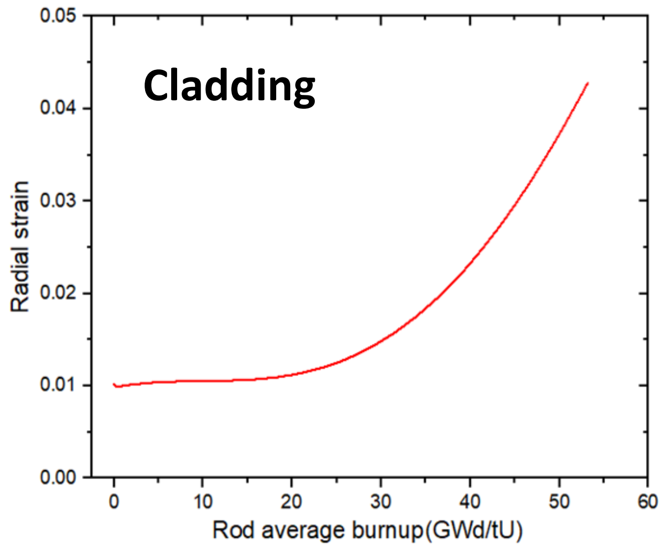
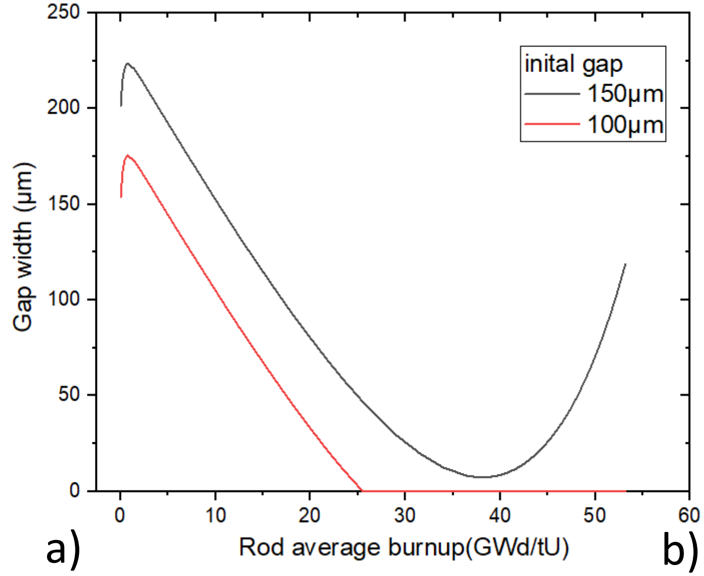
Gap size evolution

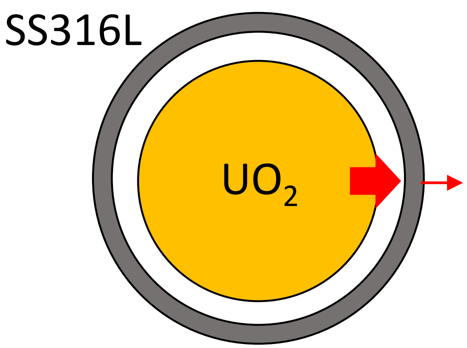




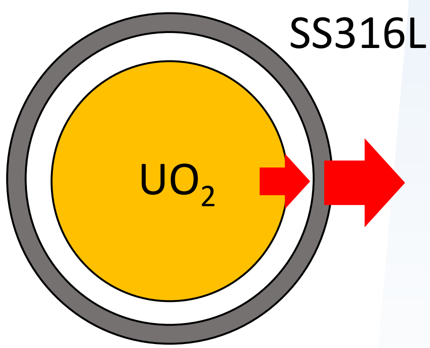
<Gap size evolution>



150 μm gap
ensure no contact

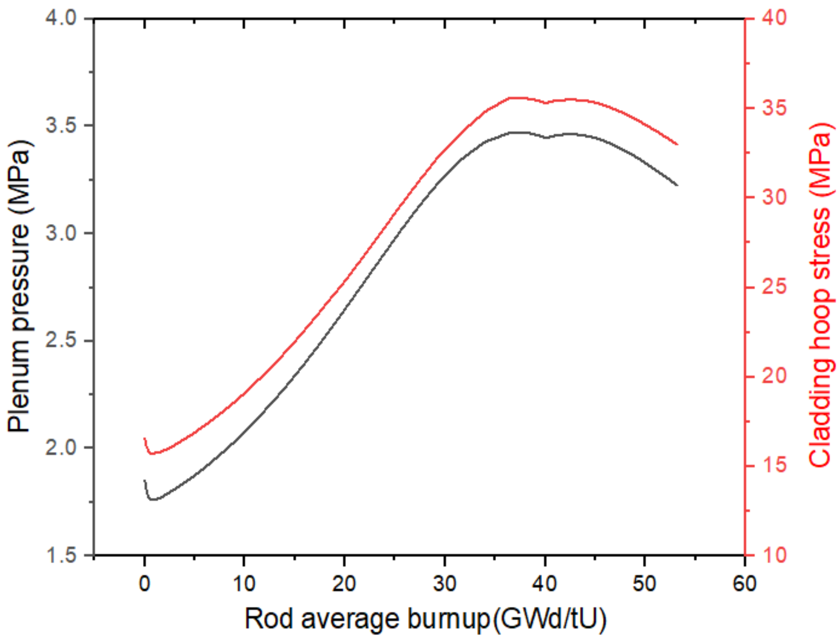


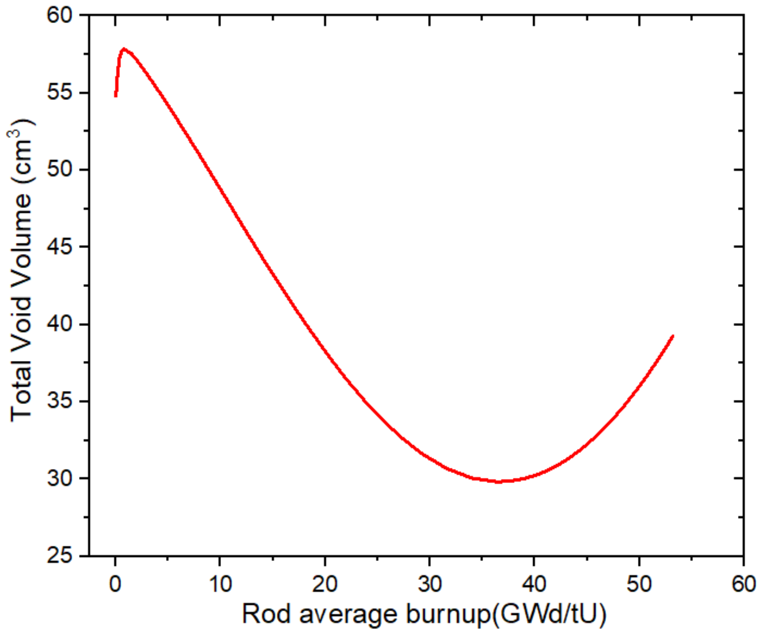
Gap decrease due to larger fuel swelling



Gap increase due to larger cladding swelling

Mechanical analysis





- Increase of plenum pressure due to void volume decrease

- After 40GWd/tU, plenum pressure decrease due to large expansion of the cladding
➔ increases the gap, total void volume

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Mechanical analysis

- Cladding integrity 10cm plenum length, 1mm cladding thickness
  - Cumulative damage fractions (CDF) should be less than 10^{-5}
  - Rupture time was calculated by LMP parameter

$$LMP = T[16.0 + log_{10}(t_R)] = 1000 \times [a + b(log_{10}(\sigma_H)) + c(log_{10}(\sigma_H))^2\ ]$$

t_R = rupture time (days), σ_H = hoop stress (MPa), T = cladding temperature (573 K)

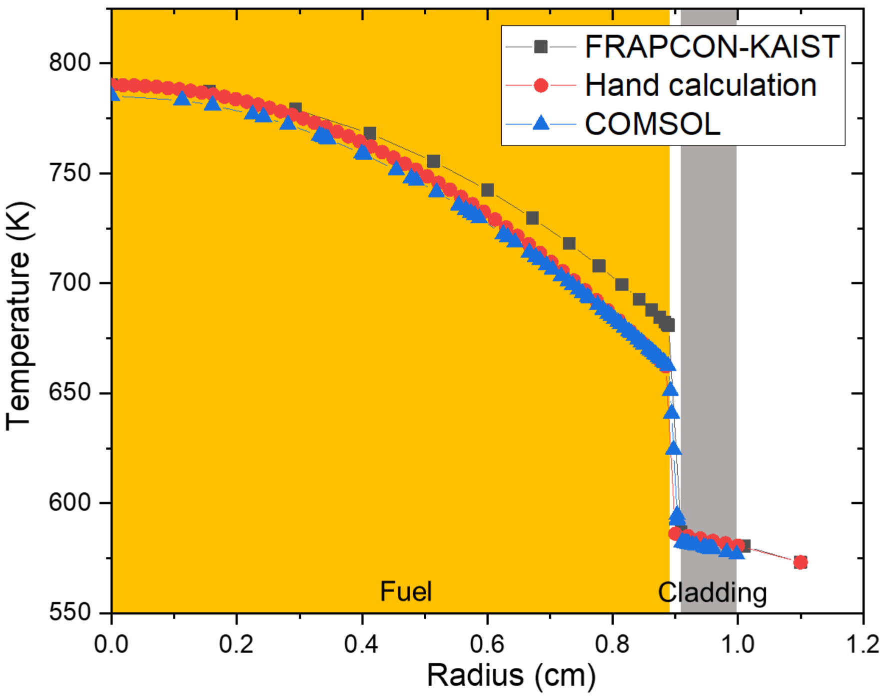
$$CDF = \int_0^t \frac{dt}{t_R}$$

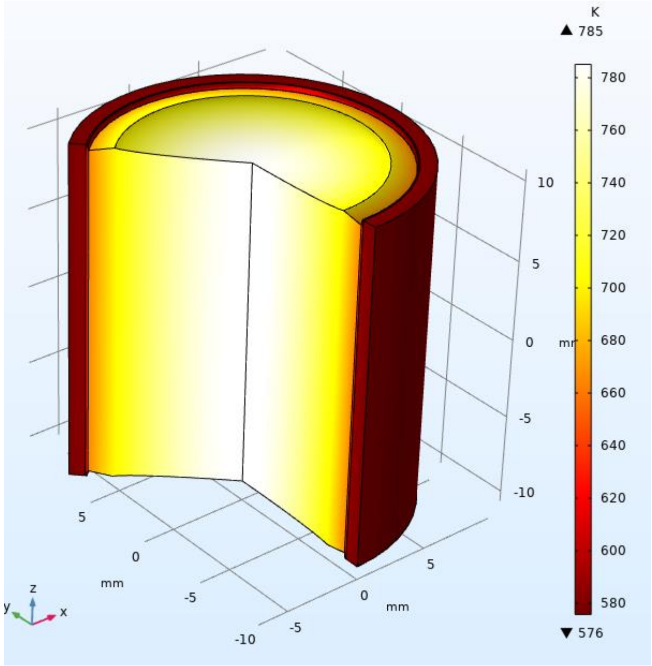
Parameters for creep ruture correlation		Modified type 316
$\sigma_H > 110\,MPa$	a	5.8640
	b	16.161
	c	−4.7730
$\sigma_H \leqq 110\,MPa$	a	25.752
	b	−3.3240
	c	0.0000

Table 5 Parameters for in-reactor creep rupture correlation

  - CDF = $3.80*10^{-16}$ (due to very low temperature, low fission gas release)

Validation





- Only thermal analysis results were cross-checked by COMSOL
- Due to the larger initial gap in FRAPCON calculation (thermal expansion at the beginning) higher pellet surface temperature in FRAPCON results.
- FRPCON simulation was in good agreement with both hand calculation and COMSOL

Conclusions

Conclusions

- **LWR fuel performance analysis code FRAPCON-4.0 was modified to be applied to micro LFR**
- **Material properties of the cladding, coolant and low temperature void swelling characteristics of the fuel were changed.**
- **It has been shown that fuel can be maintained at temperatures as low as 1000 K or less for 40 years of operation.**
- **In addition, the design of the nuclear fuel and cladding does not contact during the life time, thereby preventing cladding failure due to fuel cladding mechanical interaction (FCMI).**


Thank you