Finite Element Analysis of Unit Cell of Inverted Core Fuel for Micro Lead-cooled Fast Reactor

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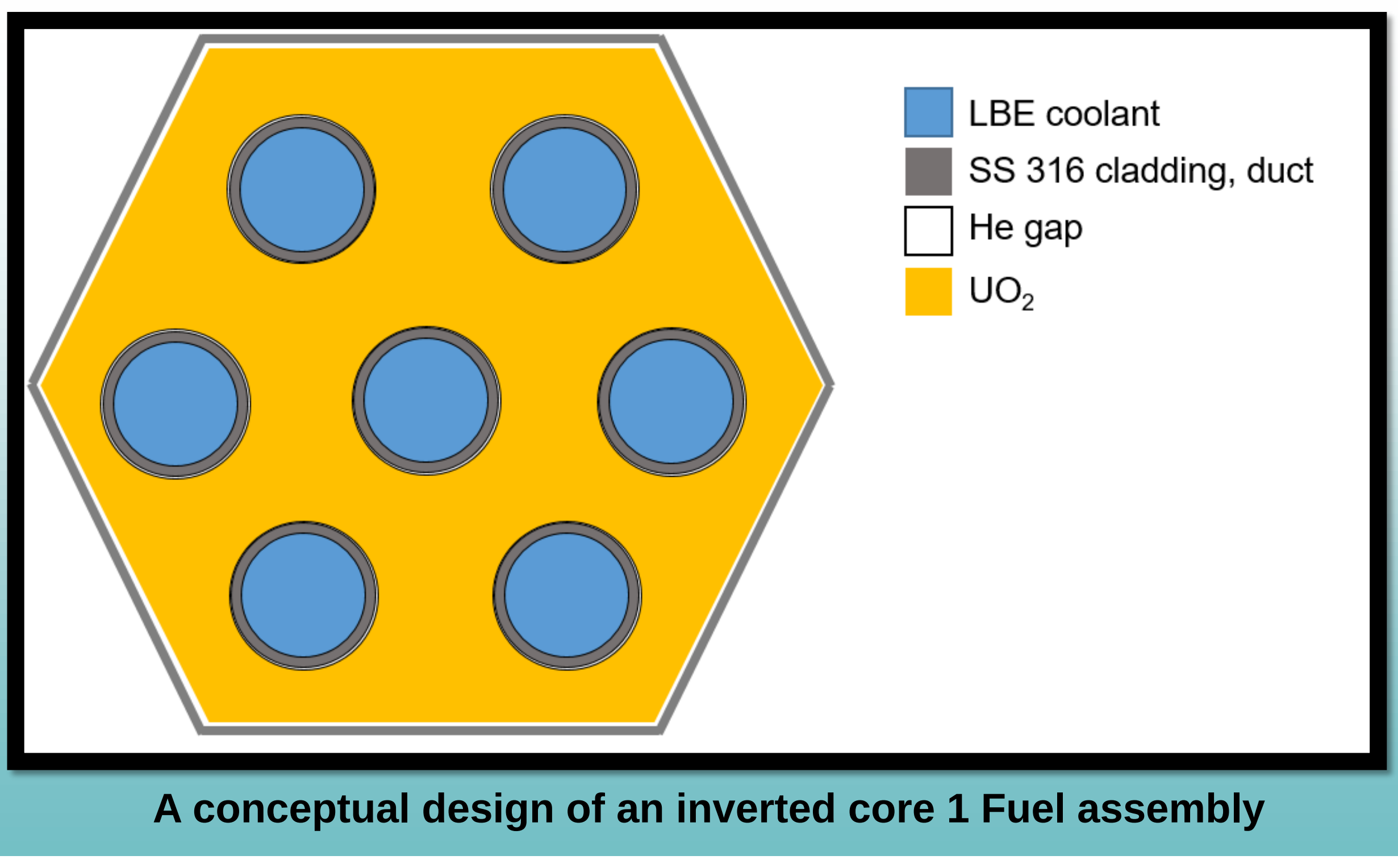
OBJECTIVES OF THE STUDY

- Evaluating the inverted fuel assembly and unit cell temperature and thermal stress with finite element analysis.

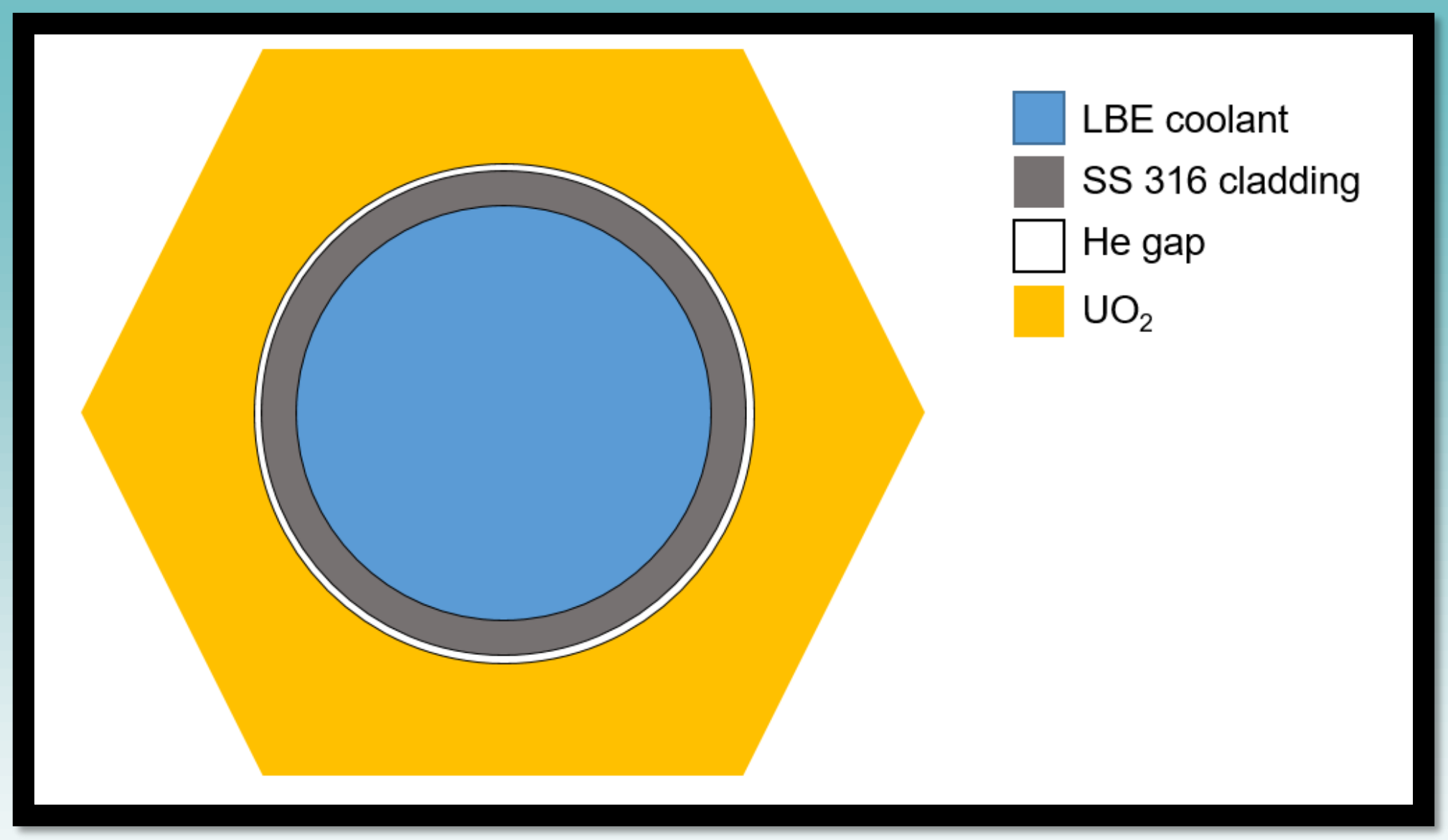
Introduction

- In the case of a marine nuclear power reactor, cladding failure due to fretting wear between the spacer grid and cladding due to the vibration of the ship is highly probable.
- To ensure the long-term mechanical integrity of nuclear fuel rods, a concept of inverted core fuel is considered.
- Coolant channel presents within the fuel in Inverted core fuel concept and fuel grid structure (grid, wire wrap..) from the coolant side can be eliminated.
- Fretting wear of cladding is not concern and it is possible to improve cladding failure resistance.



A conceptual design of an inverted core 1 Fuel assembly

Simulation Conditions

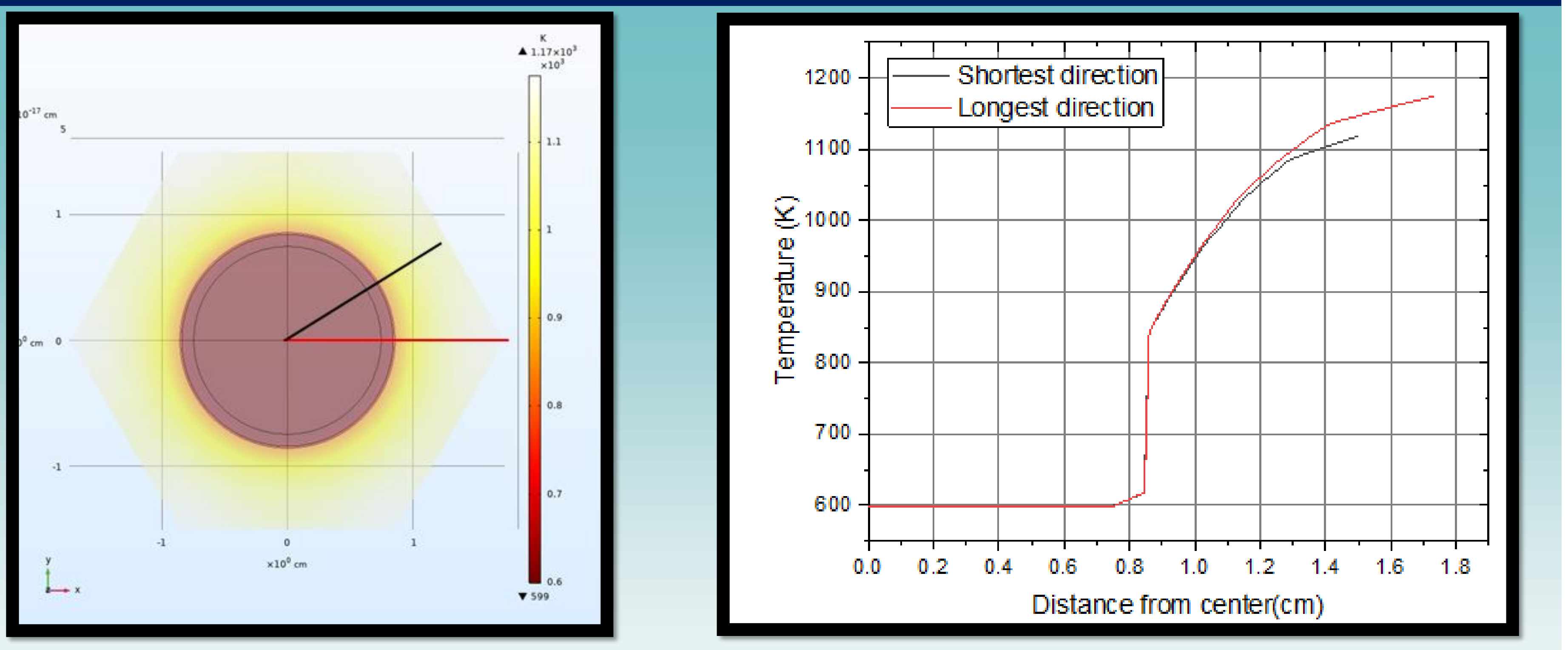


Schematic cross-section image of a unit cell inverted core fuel

Table 1. Finite analysis conditions of inverted core fuel assembly and unit cell

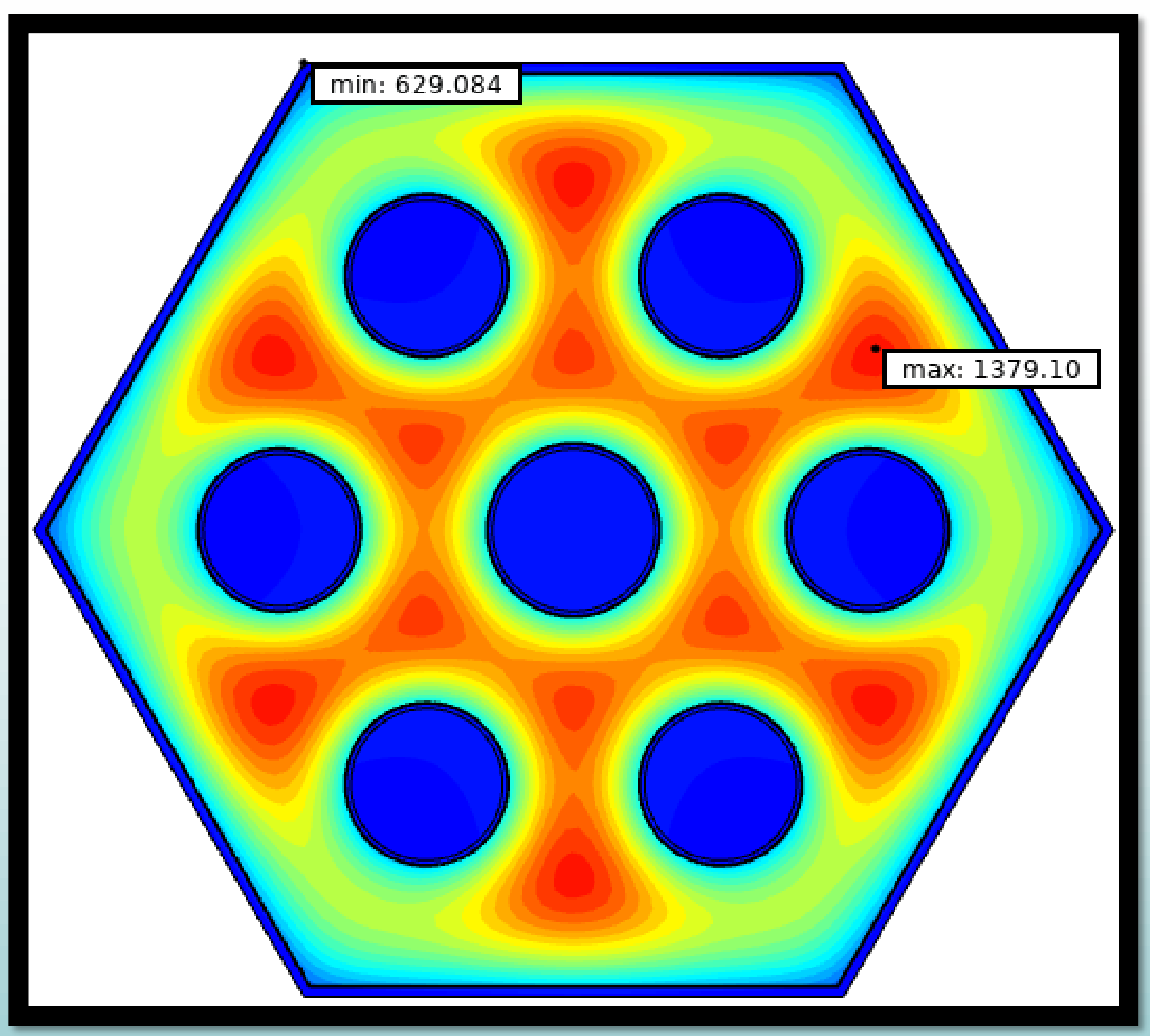
Design Factor	Design Value
Unit cell design	
Coolant channel diameter D_c (cm)	1.6
Helium gap (cm) /Cladding thickness (cm)	0.015/ 0.095
P_{fuel} (cm)	3.1
Coolant temperature (K)	623
Convective heat transfer coeff. of coolant (W/m^2K)	14509
Fuel power density (W/cm^3)	36.0
Fuel assembly design	
FA helium gap (cm)	0.015
FA wrapper (cm)	0.095
Total_FA_pitch (cm)	9.22

Temperature Analysis Results



Radial temperature distribution of unit cell of inverted core fuel

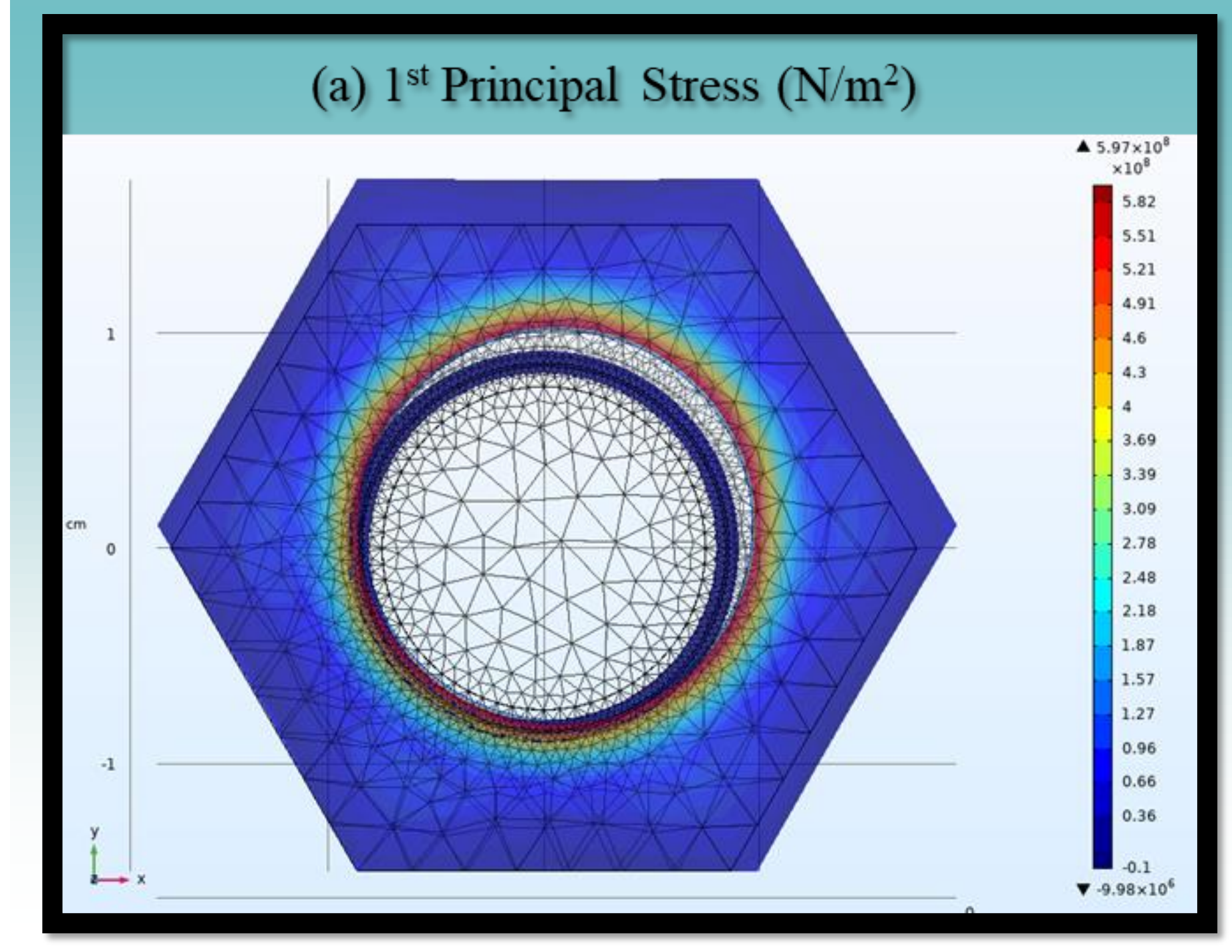
- Temperature difference occurs at both ends of the red line and the black line.
- This is because the distance from the coolant is different according to the direction in the case of the hexagonal unit cell.

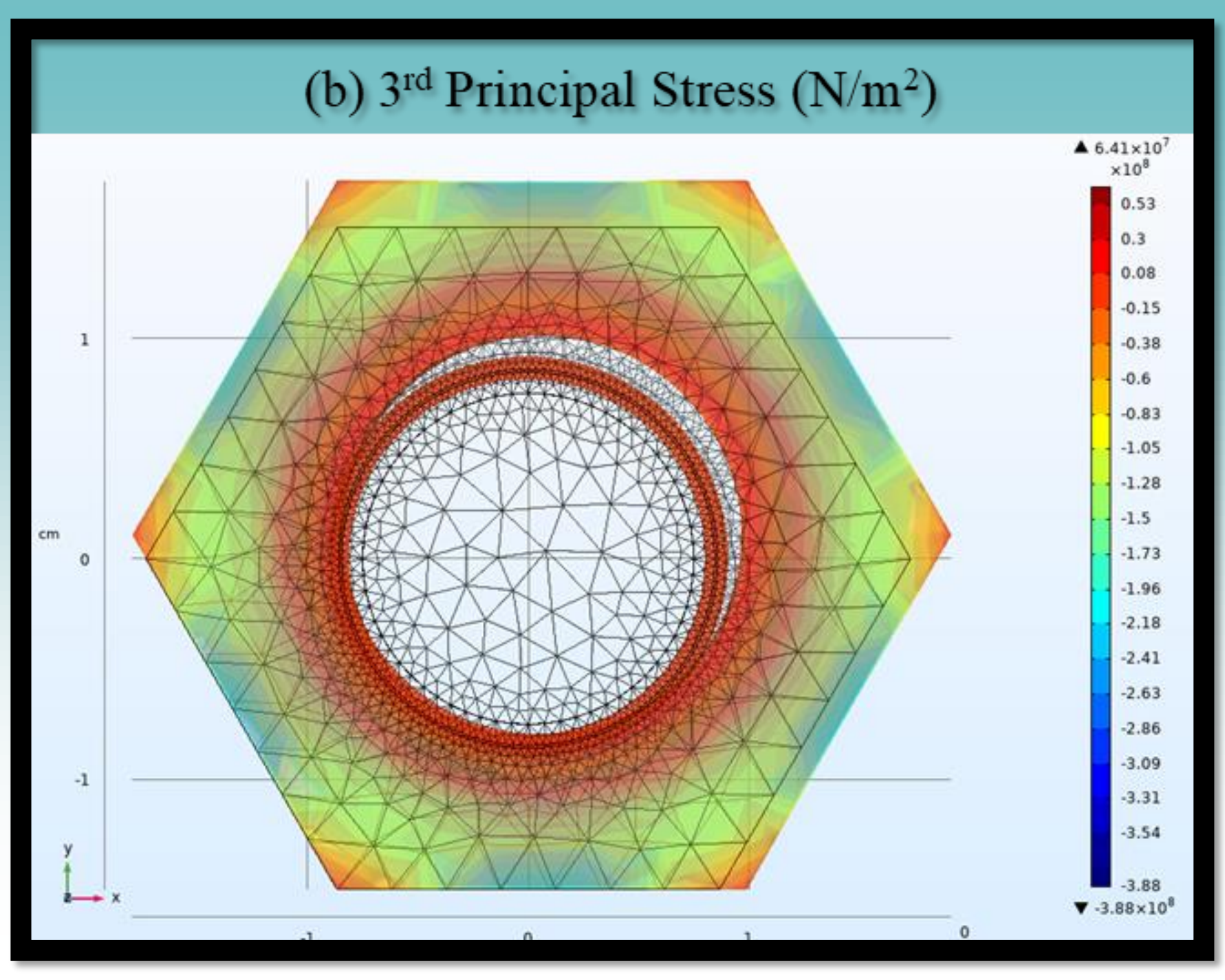


2D temperature distribution of inverted core fuel assembly

- By geometry optimization process, heat transfer analysis results shows well distributed temperature profile.

Mechanical Analysis Results


(a) 1st Principal Stress (N/m^2)


(b) 3rd Principal Stress (N/m^2)

- First principal stress, which is the tensile stress, rises to a maximum of 600 MPa at the inner surface of fuel.
- Compressive stress is calculated as a maximum of 65 MPa by this thermal gradient due to the asymmetric geometry.
- The fracture strength of UO_2 is about 200 MPa at 800℃, current stress level exceeds the fracture strength of UO_2, and a thermal crack inside the nuclear fuel will occur.

CONCLUSIONS

- In this study, the thermal and mechanical properties of unit cell and fuel assembly of inverted core fuel were evaluated by finite element analysis.
- Fuel temperature gradients arise due to the hexagonal asymmetric fuel geometry and due to the thermal gradient, there was compressive thermal stress in the corner of the hexagon unit cell.
- At the inner surface of the inverted core fuel channel, tensile stress was calculated, which exceeds the fracture strength of UO_2 at the temperature.

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Nuclear Fuel Materials Laboratory

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