POLITECNICO DI MILANO
DIPARTIMENTO DI ENERGIA

HOMEWORK 2023-24

PRELIMINARY DESIGN AND VERIFICATION OF A FUEL PIN

POLITECNICO DI MILANO

DEPARTMENT OF ENERGY - NUCLEAR ENGINEERING DIVISION

Nuclear Design and Technology

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Objective. The goal of this project is to perform the preliminary design of a fuel pin for a sodium-cooled fast reactor.

You are asked to complete the **preliminary sizing** of the fuel pin, by determining:

1. The thickness of the cladding
2. The size of the fuel-cladding gap
3. The height of the plenum

Based on the sizing step, you are asked to **verify** the preliminary pin design proposed in terms of:

1. Temperature of the cladding
2. Yielding of the cladding
3. Thermal creep of the cladding
4. Margin to melting of the fuel

The project activity is to be performed in groups.

A mid-term review is planned for November 2023.
For the mid-term review, you are asked to deliver the preliminary verification of the cladding temperature and of the margin to fuel melting.

This review is solely for self-evaluation and does not contribute to the project final evaluation.

Deliverables:
Please, send to lelio.luzzi@polimi.it:

- A technical report in .pdf (**max. 5/6 pages**) detailing the project results and conclusions. Please, specify and discuss in the .pdf report all the assumptions and approximations made throughout the design and verification steps.
- A zip folder with all the files used to perform the calculations.

Please, include your group number and all your surnames in the files name.

Due date: 12/01/2024

Summary

The goal of this project is to design the inner-core fuel pin of a sodium-cooled fast reactor. In the following, you will find information about the overall reactor core design, the material properties to be considered, and the reference design and irradiation scenario to be assumed.

The core design specifications of this Sodium-cooled Fast Reactor (SFR) are given in Table 1. The reactor core is composed of wrapped hexagonal Fuel Assemblies (FAs), each one containing 217 fuel pins arranged in a triangular lattice. The FAs are subdivided into two radial zones of the core, and surrounded by two rows of reflector and neutron-shielding elements (Figure 1).

The fuel pin concept is based on an axially heterogeneous fuel column (i.e., active column composed by fertile and fissile axial zones), with annular fissile pellets while full fertile pellets (Figure 1). The fuel is made of U-Pu mixed oxides (MOX) for what concerns the fissile pellets (initial Pu content of 23 at.% over heavy metal atoms), while the fertile pellets are made of uranium dioxide (UO_2, natural U). The cladding material is an austenitic stainless steel 15-15Ti (15% Cr, 15% Ni, 0.45% Ti). The initial filling gas in the fuel-cladding gap is 100% helium. Detailed compositions of fuel and cladding are provided in Table 2.

The irradiation scenario consists in a normal operation at the reactor power of 1500 MW_{th}. The fuel residence time in-reactor is of 1440 EFPD (Equivalent Full Power Days), divided in four cycles of 360 EFPD each, with a progressive decrease by 10% of the pin linear power during irradiation. The maximum linear heat rate of q' = 46.3 kW m^{-1} (hot pin, hot channel) occurs at the beginning of irradiation (i.e., after the first power rise at reactor start-up). The axial peak factors defining the axial profile of the pin linear power and neutron flux, assumed constant along irradiation and equal for the average and the hot channel, are provided in Table 3.

A list of correlations for the properties of the fuel, cladding and coolant is provided in Table 4. The equivalent stress leading to thermal creep failure is plotted in Figure 2 against the Larson-Miller parameter (LMP), for the considered cladding steel.

The fuel pin design limits, to be respected under normal operation irradiation conditions, are collected in Table 5.

If any additional information should be needed to complete the activity, please make consistent and justified assumptions.

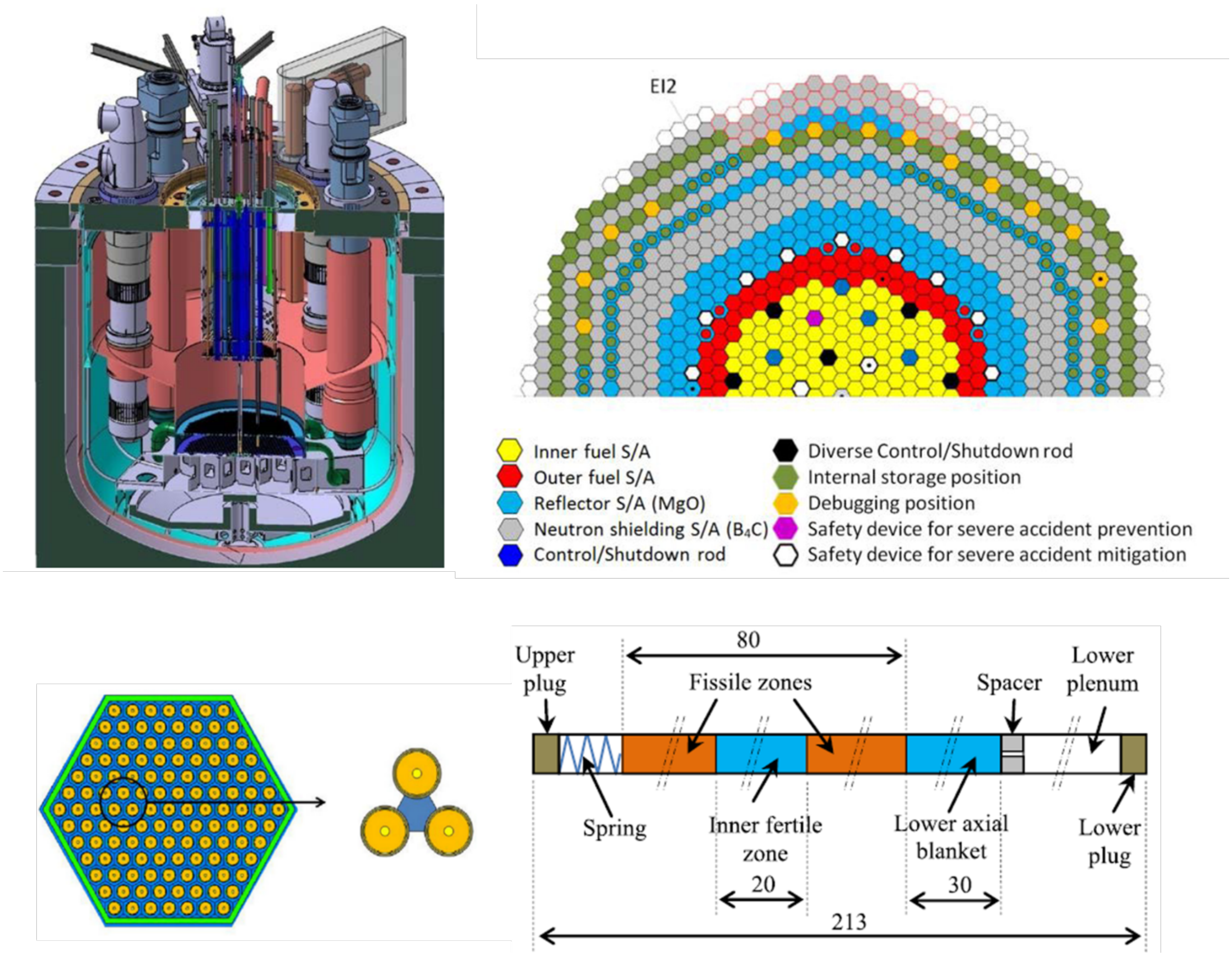

Figure 1: Reactor core sketches (pin dimensions in cm).

Table 1: Design specifications of the reactor core and as-fabricated fuel pins.

Reactor specifications	**Values**
Thermal power (MW)	1500
Number of inner-core fuel assemblies	180
Number of outer-core fuel assemblies	108
Number of pins per fuel assembly	217
Pin pitch (mm)	10.7
Coolant inlet temperature (°C)	400
Coolant inlet pressure (MPa)	0.3
Coolant mass flow rate per assembly (kg s^{-1})	22
Target fuel burnup (GWd t_{HM}^{-1})	135 (hot channel)
Total neutron flux (10^{15} n cm^{-2} s^{-1})	3.38 (hot channel)
Fast neutron flux (10^{15} n cm^{-2} s^{-1}) (> 100 keV)	70% of total neutron flux

Fuel pin specifications	**Values**
Total active length (mm)	1100
Lower fertile zone height (mm)	300
Lower fissile zone height (mm)	250
Inner fertile zone height (mm)	200
Upper fissile zone height (mm)	350
Fuel inner diameter (mm) – fissile pellets	2.45
Fuel inner diameter (mm) – fertile pellets	0
Fuel outer diameter (mm)	8.46
Cladding outer diameter (mm)	9.70
Wire spacer diameter (mm)	1
Filling gas pressure (MPa)	0.1
Filling gas temperature (°C)	20
Fuel density (% of Theoretical Density (TD))	95
Fuel grain diameter (μm)	10
Fissile pellets:	
Pu/(Pu+U) (at.%)	23
Oxygen-to-metal ratio, O/M (/)	1.97
Fertile pellets:	
Pu/(Pu+U) (at.%)	0
Oxygen-to-metal ratio, O/M (/)	2.00

Table 2: Materials composition (percentages of isotope atoms element atoms).

Fuel isotopic specifications (wt.%)	**Values**
(^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu)	(0.24, 67.87, 26.07, 4.62, 1.2)
(^{234}U, ^{235}U, ^{236}U, ^{238}U)	natural U *
Cladding composition (wt.%)	**Values**
(Cr, Ni, Mo, Mn, Si, Ti, C)	(15.0, 15.0, 1.5, 1.5, 0.9, 0.4, 0.09)
B (ppm)	60
Filling gas (%)	**Values**
He	100

* Natural U in both fertile and fissile pellets.

Table 3: Axial peak factors for both pin linear power and fast neutron flux.

Node number	Height from bottom of fuel column - cold geometry (mm)	Value
1	0	0.001
2	130	0.005
3	230	0.01
4	290	0.03
5	300	0.205
6	310	0.38
7	350	0.45
8	400	0.51
9	450	0.55
10	490	0.57
11	540	0.58
12	550	0.315
13	560	0.05
14	590	0.04
15	630	0.04
16	670	0.05
17	700	0.06
18	740	0.08
19	750	0.46
20	760	0.84
21	770	0.89
22	790	0.94
23	830	0.98
24	880	1
25	930	0.98
26	980	0.92
27	1020	0.86
28	1050	0.81
29	1070	0.77
30	1100	0.71

Table 4: Material properties.

Fuel properties	**Correlations**
Melting temperature (K) (bu in [GWd/t_{HM}])	$T_m = 2964.92 + [(3147 - 364.85 \cdot [Pu] - 1014.15 \cdot x) - 2964.92] \cdot e^{\frac{-bu}{40.43}}$
Thermal conductivity (W m^{-1} K^{-1}) – U-Pu MOX (T in [K], bu in [/ FIMA] = fractional burnup)	$\lambda = \left(\frac{1}{1.528\sqrt{x+0.0093} - 0.1055 + 0.44 \cdot bu + 2.885 \cdot 10^{-4} \cdot T} + 76.4 \cdot 10^{-12} \cdot T^3\right) \cdot 1.16 \cdot \frac{1-p}{1+2p}$
Thermal conductivity (W m^{-1} K^{-1}) – UO_{2-x} (T in [°C], bu in [/ FIMA] = fractional burnup)	$\lambda = \left(\frac{1}{0.115 + 2.6 \cdot 10^{-3} \cdot bu + 2.475 \cdot 10^{-4} \cdot T} + 1.216 \cdot 10^{-2} \cdot e^{1.867 \cdot 10^{-3} \cdot T}\right) \cdot \frac{1-(2.58 - 5.8 \cdot 10^{-4} \cdot T) \cdot p}{1-(2.58 - 5.8 \cdot 10^{-4} \cdot T) \cdot 0.05}$
Linear thermal expansion coefficient (°C^{-1}) (reference: 25°C)	$\alpha_L = 1.2 \cdot 10^{-5}$
Young's modulus (MPa)	$E = (22.43 \cdot 10^4 - 31.19 \cdot T[°C]) \cdot (1 - 2.6 \cdot p)$
Poisson coefficient (-)	$\nu = 0.32$
Swelling strain (%) (bu in [GWd/t_{HM}])	$\varepsilon_{swe,tot} = 0.07 \cdot bu$
Cladding properties	
Melting temperature (°C)	$T_m = 1400$
Linear thermal expansion (%)	$\varepsilon_{th} = -3.101 \cdot 10^{-4} + 1.545 \cdot 10^{-5} \cdot T[°C] + 2.75 \cdot 10^{-9} \cdot T[°C]^2$
Density (kg m^{-3})	$\rho = 7900 \cdot (1 + \varepsilon_{th})^{-3}$
Specific heat (J kg^{-1} K^{-1})	$c_p = 431 + 0.77 \cdot T[K] + 8.72 \cdot 10^{-5} \cdot T[K]^{-2}$
Thermal conductivity (W m^{-1} K^{-1})	$\lambda = 13.95 + 0.01163 \cdot T[°C]$
Young's modulus (GPa)	$E = 202.7 - 81.67 \cdot 10^{-3} \cdot T[°C]$
Poisson coefficient (-)	$\nu = 0.277 + 6 \cdot 10^{-5} \cdot T[°C]$
Yield stress (MPa) at 0.2% strain	$\sigma_{Y,0.2\%} \begin{cases} 555.5 - 0.25 \cdot T[°C] & \text{if } T < 600°C \\ 405.5 - 0.775 \cdot (T[°C] - 600) & \text{if } 600°C < T < 1000°C \\ 345.5 - 0.25 \cdot T[°C] & \text{if } T > 1000°C \end{cases}$
Ultimate tensile strength (UTS) (MPa)	$\sigma_{UTS} \begin{cases} 700 - 0.3125 \cdot T[°C] & \text{if } T < 600°C \\ 512.5 - 0.969 \cdot (T[°C] - 600) & \text{if } 600°C < T < 1000°C \\ 437.5 - 0.3125 \cdot T[°C] & \text{if } T > 1000°C \end{cases}$
Rupture strain (%)	$\varepsilon_{rupt} = 8 + 4.74 \cdot 10^{-3} \cdot (T[°C] - 500) + 6.2 \cdot 10^{-5} \cdot (T[°C] - 500)^2$
Void swelling (%)	$\frac{\Delta V}{V} = 1.5 \cdot 10^{-3} \exp\left[-2.5\left(\frac{T[°C] - 450}{100}\right)^2\right]\left(\frac{\varphi}{10^{22}}\right)^{2.75}$
Thermal creep strain rate (% h^{-1})	$\dot{\varepsilon_\vartheta} = 2.3 \cdot 10^{14} \exp\left(\frac{-84600}{R \cdot T[K]}\right) \sinh\left(\frac{34.54 \cdot \sigma_{eq}}{0.8075 \cdot R \cdot T[K]}\right)$
Irradiation creep strain rate (% h^{-1})	$\dot{\varepsilon_{irr}} = 3.2 \cdot 10^{-24} \cdot \overline{E} \cdot \phi \cdot \sigma_{eq}$
Larson-Miller parameter (LMP)	$LMP = \begin{cases} T[K] \cdot (17.125 + \log_{10}(t_{rupt})) \\ \frac{2060 - \sigma_{eq}}{0.095} \end{cases}$
Coolant properties	
Melting temperature (°C) at atmospheric pressure	$T_m = 98$
Boiling temperature (°C) at atmospheric pressure	$T_b = 882$
Specific heat (J kg^{-1} K^{-1})	$c_p = 971.34 - 3.69 \cdot 10^{-1} \cdot T[°C] + 3.43 \cdot 10^{-4} \cdot T[°C]^2$
Density (kg m^{-3})	$\rho = 954.1579 + T[°F] \cdot (T[°F] \cdot (T[°F] \cdot 0.9667 \cdot 10^{-9} - 0.46 \cdot 10^{-5}) - 0.1273534)$
Dynamic viscosity (Pa s)	$\eta = 10^{-3} \exp\left(2.3 \cdot (0.5108 + \frac{220.65}{T[K]} - 0.2139 \cdot \log(T))\right)$
Thermal conductivity (W m^{-1} K^{-1})	$\lambda = 94 - 3.25 \cdot 10^{-2} \cdot T[°F] + 3.62 \cdot 10^{-6} \cdot T[°F]^2$
Nusselt number (-)	$Nu = 7 + 0.025 \cdot Pe^{0.8}$
Filling gas Thermal conductivity (W m^{-1} K^{-1})	$\lambda_{He} = 15.8 \cdot 10^{-4} \cdot T[K]^{0.79}$

Variables: bu = burnup (/ FIMA or GWd/t_{HM}); $\overline{E}$ = mean neutron energy (MeV); p: porosity (/TD); [Pu]: Pu content (at.%); R = gas constant = 1.986 cal mol^{-1} K^{-1}; T: temperature (°C or K or °F); t_{rupt} = time-to-rupture (h); x = 2.00-O/M: deviation from stoichiometry (-); φ = neutron fluence (n cm^{-2}); ϕ: neutron flux (n cm^{-2} s^{-1}); σ_{eq} = Von Mises equivalent stress (MPa).

Table 5: Indicative design limits.

Quantity	Design limits
Maximum fuel temperature	< 2400°C
Maximum cladding outer temperature	< 620°C (at cladding mid-wall)
Maximum plenum pressure	< 5 MPa
Instantaneous cladding plastic strain	< 0.5%
Cladding thermal creep strain	< 0.2%
Cladding volumetric swelling	< 6%

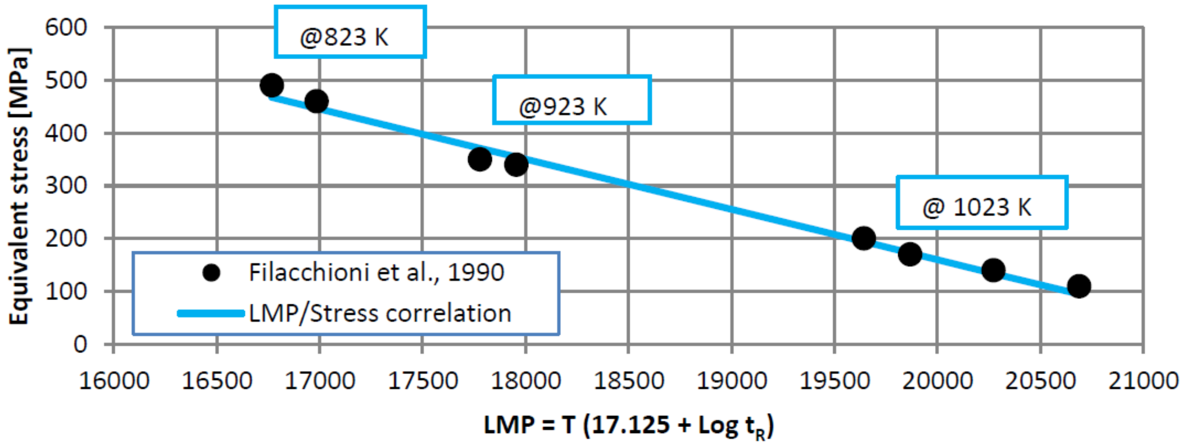

Figure 2: Equivalent stress leading to creep failure vs. Larson-Miller parameter (LMP) for the considered cladding steel (T: temperature in K, t_R: time-to-rupture in hours).