Homework
Nuclear Design and Technology
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1 Introduction and description of the project

The goal of this project is to perform the preliminary design of a fuel pin for a lead-cooled fast reactor. You are asked to complete the **preliminary sizing** of the fuel pin, by determining:

- the thickness of the cladding;
- the size of the fuel-cladding gap;
- the height of the plenum.

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

- outer temperature of the cladding;
- yielding of the cladding;
- thermal creep of the cladding;
- margin to melting of the fuel.

2 Cold Geometry

The modeling and sizing steps can be broken down into:

- With some preliminary calculations we found the mass flow rate ($\Gamma = 0.59\,[kg/s]$), then we derived the profiles of thermal power in average and hot conditions and the relative maximum values, using the given axial peak factor table and mantaining the profiles discretized along the z direction, with 24 steps each.
- The next step consisted in obtaining the radial temperature profile, from the bulk to the inner fuel radius. First of all, to find the convective heat transfer coefficient h, we adopted the most conservative of the two correlations for Nusselt number (Nu) mentioned in the Handouts paper:

$$Nu = 4.5 + 0.014 \cdot Pe(D_h, T)^{0.8} \tag{1}$$

  Were D_h is the hydraulic diameter and T the temperature.
- We calculated the profile of the bulk coolant temperature implementing a for cycle on python in order to model the temperature at each step of the z axis.

$$T(z) = T_{in} + \frac{Fq'(z)L'}{\pi\Gamma c_p} \tag{2}$$

  We verified that the T of the coolant at the outlet in average conditions is $480°C$ as expected. For the hot channel we found instead $494°C$.
- To find the outer coating temperature profile we integrated the Newton equation for convection , using a cycle to find the right values of the convective heat exchange coefficient for each z section.

$$q'' = h\Delta T \tag{3}$$

- By applying Fourier's equation for conductive heat transfer we were able to obtain also the inner coating T.
- We continued the analysis of the radial temperature profile in the cladding, the gap and the fuel. Considering cylindrical coordinates and equilibrium configuration, we were able to find the profile of the inner cladding temperature by integrating the Fourier's equation:

$$\rho c_p \frac{dT}{dt} = \nabla \cdot K\nabla T + q''' \tag{4}$$

For both cleaning and gap we derived the following:

$$\int_{T_{out}}^{T_{in}} kdT = \frac{q'}{2\pi} \ln \frac{r_{out}}{r_{in}} \tag{5}$$

whereas for the fuel we obtained:

$$\int_{T_{out}}^{T_{in}} kdT = \frac{q'}{4\pi} F_v \tag{6}$$

In particular we considered one step of z at a time and for each we calculated the thermal conduction coefficient by integrating it and we implemented a cycle to find the temperature with the constriction that this integral must be equal to the one calculated of the linear power depending on the radius.

- The same procedure was followed to find the temperature profiles at the outer and inner fuel radii. In this way we were able to obtain plots of the various axial temperature profiles in the various zones and evaluate the maximum temperatures of each, as we can see in Figure 1. We concluded that with a gap thickness of $110\mu m$, the maximum temperature of the inner fuel in hot channel condition respected the design limits for maximum fuel temperature, because it was $1994°C$ ($< 2000°C$). Regarding instead the safety limit on the outer cladding temperature, which imposed $T_{co} < 550°C$, we found a maximum temperature of $547°C$ for the hot channel, which is in line with the expected results. Thus with a gap of $110\mu m$ the resultant cladding thickness is $0.64mm$.

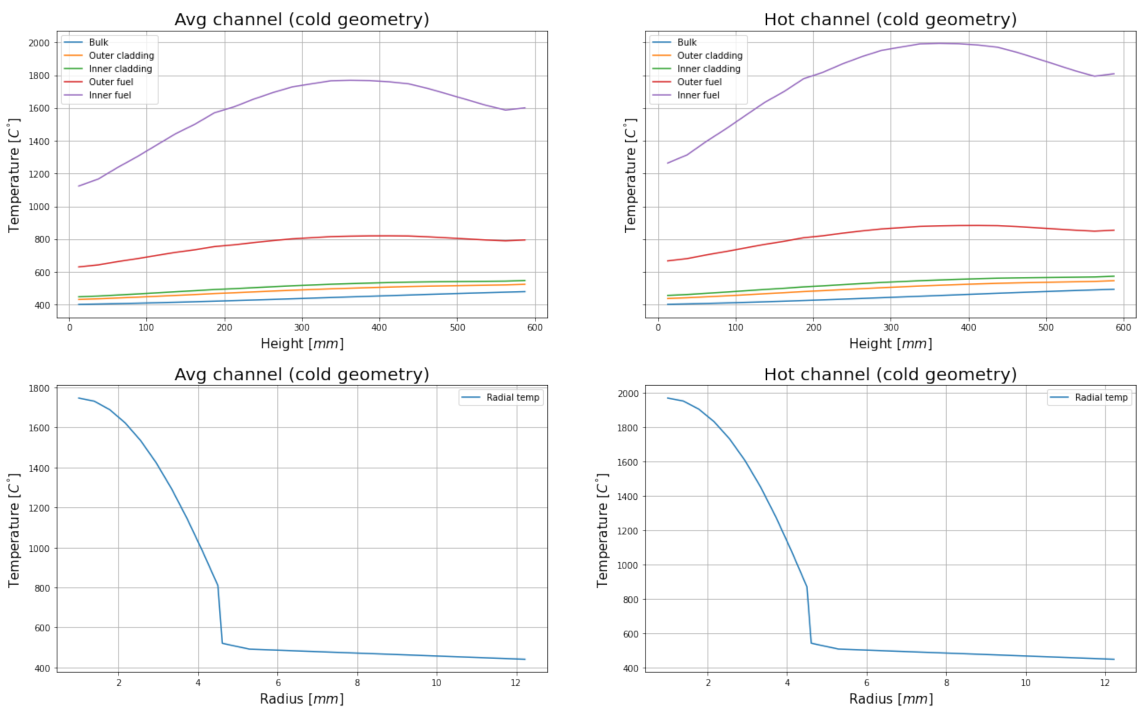

Figure 1: Axial and radial temperature profiles in the average and hot channel.

3 Hot geometry

The fuel is irradiated for 5-cycles at constant power without reshuffling, for a total of 5 years fuel residence time in reactor and a total $EFPD = 5 \cdot 365 = 1825$ days.

We started analyzing the restructuring of the fuel that happens at the beginning of life. By considering a three-zone division of the fuel (the central void should also be considered):

- As fabricated;
- equiaxed grains;
- columnar grains.

Applying the mass conservation equations, finding k in each zone by integrating between the temperatures at the edges of each zone we could evaluate the restructuring in the MOX fuel, obtaining $r_{columnar}$ and $r_{equiaxed}$. Only in hot channel significant changes were noticed. It should be kept in consideration that the inner radius grows to balance the densification process. Figure 2 displays the restructuring at zero irradiation.

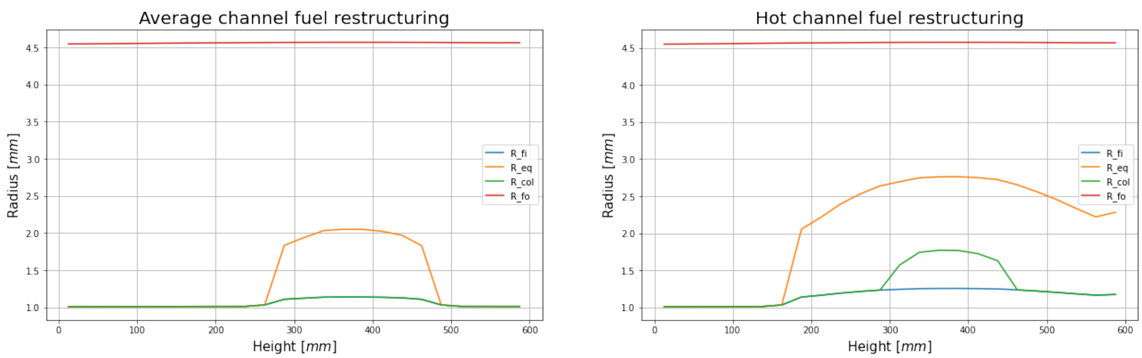

Figure 2: Axial restructuring in the fuel for the average and hot channel.

- After the restructuring one of the requests of the assignment was to size the height of the plenum, to do this one must regard the fission gas release and He release.
  We begin by considering the fission gas concentration
  $$\frac{dP}{dt} = y\dot{F} \tag{7}$$
  where $\dot{F}$ is the fission rate and y is the yield of the various gases. We considered as yields 0.27 for Xe, 0.03 for Kr and for He we used the values found in Akie's et al.[1].
  We integrated the equation in time using a $EFPD$ of 1825. We considered an ideal gas behaviour for both the released gases and the Helium already present in the plenum. Putting together the two equations, with pressure is the limit pressure (5MPa), and temperature equal to the inner cladding one (hottest point). we were able to find a plenum height of $h_{plenum} = 0,52m$, which is in line with our expectations. We assumed 100% of gas release to be conservative, the actual value is lower.
- The following step we took for hot geometry was the thermo-mechanical analysis. We employed the solutions of the Pipe Equation to find the three thermal stresses (σ_r,σ_z,σ_θ), both for the fuel and the cladding. The hypothesis made to apply the Pipe equation are axial-symmetry, orthocylindricity and homogeneneous and isotropic material. For what concerns the mechanical stresses we used the Lamè solutions with boundary conditions: external pressure is the coolant's one, assumed to be around $P_{cool} = 0.5$MPa; internal pressure is the one of the gap, if the gap is opened, which is $P_{gap} = 5$MPa at the worst condition, while it is the contact pressure if the gap is closed. The equation of the contact pressure was found in the work by Cumo [2] We summed the thermal and mechanical part to have the three complete stresses.
- Yielding of cladding verification: we started at the beginning of irradiation cycle, and calculated the limit yielding stress from the formula given in the paper, considering as temperature the cladding highest temperature (on the hot channel), obtaining a value of 430MPa. We used the Tresca failure criterion to check if the stresses found were allowable, indeed, at the beginning of irradiation, the NDTT of steels is below $0°C$, thus we are in a ductile regime in the cladding.

- From this point we calculated the strains, accounting not only for the ones caused by the thermal + mechanical stresses, but also from thermal expansion, thermal creep, irradiation creep (for cladding only) and swelling (both for cladding (void) and fuel). We used the correlations reported in the data to find these. Finally we derived the displacements, so to evaluate the new changed geometry. This allowed us both to see the reduction of the gap and, by rerunning the thermal analysis, to find the new temperature profiles. Figure 3 shows the stresses in fuel and cladding after 5 years of irradiation. We do not show the mechanical stresses in the fuel because they are negligible (if we consider null the contact pressure).

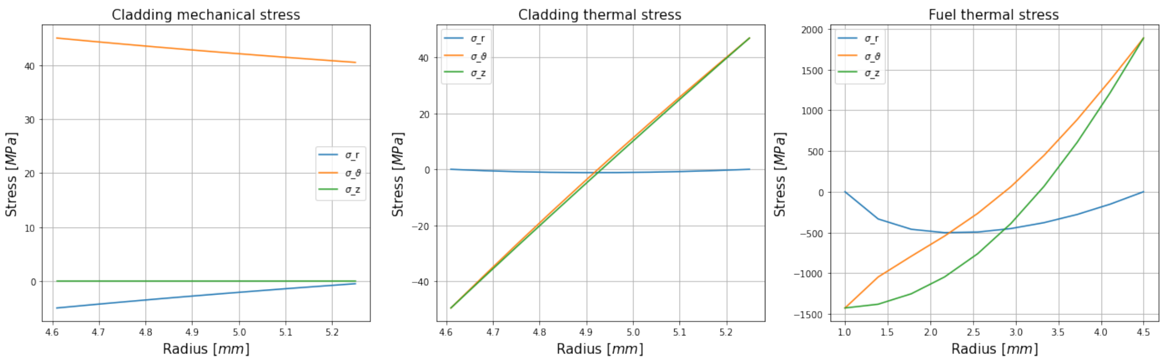

Figure 3: Mechanical and thermal stresses for fuel and cladding, evaluated radially.

In Figure 4 we plotted a graph to compare the strains due to the different causes, we saw that for short operational times, the most significant ones were the thermal expansion both for fuel and cladding plus the fuel swelling. Increasing the time span, we noticed how in the cladding the void swelling became more significant, especially where T is lower.

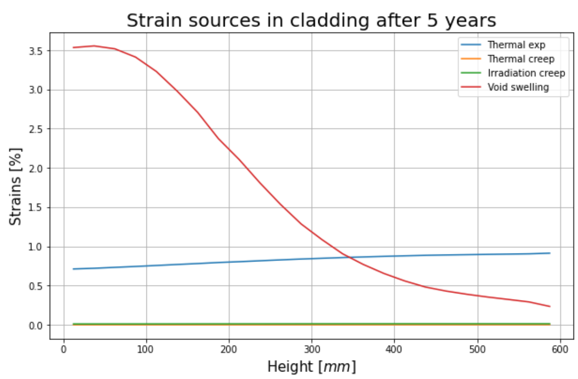

Figure 4: Axial strain evaluation after 5 years

- To check if the creep would cause failure in the operation lifetime of the reactor, we employed the Larson Miller Parameter (LMP). The time to rupture found in this way was well above the 5 years we needed.
- Finally we analyzed the embrittlement caused by Helium in the cladding. The production of He is dictated by the decay of other isotopes present in the cladding. Putting together the differential equation which describe the decays of Cr, B, Ni58, Ni59, Fe and He, the result was, after 5 years of operation, the presence of circa 2420ppm of Helium, which is above the limits we were expecting. This high value would lead to a dangerous loss of ductility of our stainless steel although the fact that we are below the reference temperature of 0.5Tm could mitigate this effect (in the worst case we found 0.4Tm). The high production of Helium could be explained with the time of irradiation considered and looking at the composition of our cladding in AIM1 stainless steel: a Manganese-Nickel austenitic one with an higher content of Cr, Ni and Fe than in ferritic.

4 Conclusions

To conclude, we report in Table 1 the design limits requested and the respective values we obtained with the code.

Comparison of results vs design limits		
Quantity	Design limit	Result
Maximum fuel temperature	2000°C	1997°C (in cold geometry)
Maximum cladding outer temperature	550°C	547°C
Maximum plenum pressure	5 MPa	5MPa
Cladding external diameter relative variation	3%	2.9% with peaks of 3.8%
Cladding thermal creep strain	0.2%	0.001%
Cladding total creep strain	3%	0.013%
Cladding swelling strain	5%	3.5%

Table 1

Finally, we want to give some remarks:

- We start observing closure of the gap in the hot channel after 1.3 years of operation. After 1.5 years the closure is completed.
- As time passes, the maximum temperature in the fuel greatly decreases due to reduction and closure of the gap, passing from 2000°C to 1700°C.
- The thermal creep we have obtained is quite low. Due to this, the calculation of the time-to-rupture with the Larson-Miller parameter is obviously too optimistic, predicting a rupture time of thousands of years. We believe this could be caused by a low intensity of the cladding stresses, since we neglected the contact pressure with the fuel.
- Although it was not specifically requested, we tried to evaluate the presence of a contact pressure between cladding and fuel after the closure of the gap. In doing so we stumbled upon some difficulties. In fact, when the contact pressure was too great ($> 80MPa$), the thermal creep strain rate predicted by the given correlation diverged exponentially. The consequence was that the cladding strains became so big that the internal radius greatly overcame the outer one and the code crashed. We tried to deal with this issue by imposing a artificial upper limit of 0.1% to the thermal creep. With this limitation we could run the code and obtain, after 5 years, values of the contact pressure up to $320MPa$, which induces (along with a higher irradiation creep of about 0.5%) a $\sigma_\theta > 2GPa$, by far bigger that the yielding and the ultimate tensile stress. In order to prevent the collapse of the cladding we tried to increase the gap, so that the gap closure would happen later and with a lower contact pressure. We found that, in order to keep the equivalent stress under the ultimate tensile stress limit, the gap must be at least $320\mu m$ thick. Of course this means having an increased inner fuel temperature ($\simeq$ 2400°C in cold geometry), which exceeds the reasonable safety limits. Overall, it is clear that this final part of the code must be improved to ensure reliable predictions.

References

[1] Hiroshi Akie, Isamu Sato, Motoe Suzuki, Hiroyuki Serizawa, and Yasuo Arai. Simple formula to evaluate helium production amount in fast reactor ma-containing mox fuel and its accuracy. *Journal of Nuclear Science and Technology*, 50(1):107–121, 2013.

[2] Maurizio Luigi Cumo. Nuclear Plants. https://www.editricesapienza.it/sites/default/files/5057_Cumo_Nuclear_Plants_eBook_offprint_0.pdf, 2017. ISBN: 978-88-9377-024-8.