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Pre-design of an AmBB pin loaded with 10% of americium

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Abstract

Minor Actinide Bearing Blanket (MABB) fuels, where minor actinides are diluted with UO_2 into the radial breeder blankets of Sodium-cooled Fast Reactors (SFR), are an option of interest for the transmutation of minor actinides in this type of reactors. However, their operation under very specific conditions raises many questions, such as the impact of the high helium production during irradiation, which, combined with relatively low temperature could enhance the fuel gaseous swelling and fuel-cladding mechanical interaction. This type of behaviour would require specific design changes in order to meet the design criteria.

This paper describes the preliminary thermo-mechanical design of an Americium Bearing Blanket (AmBB) pin and thermal-hydraulic features of an AmBB sub-assembly with the following conditions: 10% Am in UO_2 matrix, pin and fuel sub-assembly geometry of the typical ASTRID outer-core, and irradiation duration of 1940 EFPD. The behaviour of the pin is calculated with GERMINAL fuel performance code, which has been upgraded according to the specificity of AmBB fuel, with, in particular, a specific model to take into account the swelling due to gas retention. The calculation results show that thermal and mechanical margins are high, from both the design criteria (creep usage fraction and swelling of the cladding) and the thermal-hydraulic criteria (flow rate and pressure loss). Uncertainties taken into consideration do not affect this conclusion.

Keywords

Americium Bearing Blanket (AmBB) fuels, Sodium-cooled Fast Reactor (SFR), pin pre-design, thermo-mechanical design criteria, thermal-hydraulic criteria.

I. Introduction

High activity wastes are currently vitrified and planned to be stored in deep geological repositories. In order to reduce the radiotoxic inventory of vitrified wastes and the footprint of deep storage [1], research concerning solutions that could separate the most radiotoxic and long-lived elements from spent fuel and transmute them into non-radioactive or short-lived ones in nuclear reactors is being carried out on an international level. Transmutation being only reasonably applicable for Minor Actinides (MAs), (chiefly americium, neptunium, and curium) and the best transmutation performance being obtained in fast neutron reactors, MA incorporation into the fuel has become a prerequisite for Generation IV reactors to bring benefits in the disposal requirements by reducing the MA content in the high activity wastes [2]-[4]. Based on historical experience and knowledge, oxide fuels have emerged in Europe as the shorter term solution to meet the Generation IV assigned performances and reliability goals. Two main MA-recycling options have been under consideration:

- the homogeneous recycling mode, or Minor Actinide Driver Fuel (MADF) concept, where MAs are diluted in (U,Pu)O₂ standard driver fuel at a low enough content (<3%) to limit the MA impact on the performance of the fuel and on the core safety as well as on the fuel cycle facilities, as far as possible;
- the heterogeneous recycling mode on UO₂ fuel located in radial core blankets, or Minor Actinide Bearing Blanket (MABB) concept, where MAs are concentrated in UO₂ based fuels at a content of ~10% into the radial breeder blankets of Sodium-cooled Fast Reactors (SFRs) core. In this concept, leakage neutrons allow MA transmutation. Due to the lower flux in these radial locations, heterogeneous recycling is inherently less efficient than homogeneous recycling in the internal core. However, its main advantage is a low impact on operating parameters and core safety [5]. In addition, the use of the UO₂ matrix as a support for MAs should ease developments, as UO₂ behaviour under irradiation as well as UO₂ reprocessing are well known, even if MA addition during fuel manufacturing will have to be optimized.

At the beginning of investigations, experimental data on MABB fuels were scarce, with the unique irradiation SUPERFACT, performed in the PHENIX reactor in the 80's [6], [7]. In this experiment, the irradiation of U_{0.6}Am_{0.2}Np_{0.2}O_{1.926} pellets irradiated at rather high linear heat rate led to a complete release of helium during irradiation and a highly porous fuel microstructure, consistent with high temperature operating conditions (calculated to be between 1500°C to 1900°C, not fully representative of MABB operating conditions). In addition, the occurrence of a weak Fuel-Cladding Mechanical Interaction (FCMI) and the absence of a central hole were noticed.

Complementarily, a comprehensive R&D program of MABB fuel qualification started in 2008, including, as a first stage, two separate-effect irradiation tests: MARIOS [8], irradiated within FAIRFUELS [9] and examined within PELGRIMM [10] FP7 European projects, and DIAMINO [11], implemented within the French national nuclear program. They were completed by MARINE [12], the first semi-integral irradiation of AmBB fuels, carried out within PELGRIMM FP7 European project.

In parallel of these experimental tests, in 2008, a first thermo-mechanical and thermal-hydraulic pre-design of a MABB pin loaded with 20% of MA on a UO₂ matrix was performed [13]. From a thermo-mechanical point of view, the main results of this first study were:

- a high amount of produced and released helium leading to high pressurization in the pin: the criterion on the cladding primary membrane stress was dimensioning,
- to respect this criterion, and decrease the pin pressure, the plena heights had to be significantly increased.

Consequently, the resulting volume fraction of MABB fuel in the core (as a percentage of fuel in the neutronic mesh, considering a 100% density fuel) was calculated to be between 28% to 38% (instead of 41% in the neutronics scenario studies), so that the effective performances of the system had to be re-assessed by neutronics studies.

Finally, along with other transmutation studies (neutronics, fuel...), the analysis of this first pre-design study [13], led to consider the following new scenario:

- only americium transmutation to be considered as a first stage (AmBB fuel instead of MABB fuel),
- reduction of the Am content down to 10% of heavy atoms (initially 20%),
- reduction of the irradiation duration down to 1940 EPFD (initially 4100 EPFD),
- no 180-degree rotation of the MABB sub-assembly at mid-irradiation.

The objective of this paper is to update the first MABB pin pre-design study with these new considerations. The first part presents the pin modelling implemented in the fuel performance code GERMINAL V2.2 that has been adapted to the MABB specificities. The second part comments the calculation assumptions and the third part provides and discusses the results.

II. Modelling and simulation

GERMINAL V2.2 [14], part of the PLEIADES platform [15], is CEA's fuel performance code dedicated to the simulation of the irradiation behaviour of SFR (U,Pu)O₂ mixed oxide fuel pins. The simulation of transmutation fuels in heterogeneous mode ((U,Am)O₂ fuels), with a significant content of americium, requires adaptations of the models implemented in GERMINAL. The combined effect of high helium production, coming from alpha decays of MA isotopes formed from americium, with low temperatures, due to the use of a depleted UO₂ matrix (no or few fissile isotopes), can potentially lead to high gaseous swelling related to high retention of helium and fission gases at low temperatures. The specific model and laws lastly implemented in GERMINAL and their verification are presented in the first paragraph of this section. The second paragraph describes the simulation parameters constituting the reference case for the pre-design study of the AmBB pin.

II.1. Specific models and laws

II.1.1 Gaseous swelling

The main evolution implemented in GERMINAL V2.2 concerns a gaseous swelling model correlated to the high helium and fission gas retentions. This model aims to provide an "envelope" estimate of the gaseous swelling, which maximizes the pellet-cladding mechanical interaction resulting from the closure of the gap caused by the swelling of the pellet. This conservative approach is well adapted for a pre-design type study.

The gaseous swelling of the (U,Am)O₂ fuel is taken into account by the mechanical model as a stress-free strain, in addition to that induced by the solid swelling. The new gaseous swelling model implemented in the AmBB calculation scheme is largely inspired by the one describing the swelling of (U,Pu)O₂ mixed oxide fuels in transient regime [16]. The main interest of this model is that it correlates the local maximum gaseous swelling to local retention of fission gases.

For standard (U,Pu)O₂ fuel pins and their high operating conditions, the assumption commonly adopted is that the produced helium is completely released. This assumption is well founded and verified by post-irradiation examinations for fissile pins, operating at high temperatures, but for AmBB pins operating at lower temperatures, it is no longer appropriate and it is necessary to estimate helium retention. The solution to assess the contribution of helium to gaseous swelling is to apply the fission gas (xenon, krypton) retention rate to the amount of produced helium. Thus, the helium contributing to gaseous swelling is that produced in the pellet regions where the fission gases are retained. This assumption is conservative, thus consistent with a pre-design approach, because it leads to maximize the amount of helium contributing to gaseous swelling, since the fission gas retention rate maximizes that of helium.

In addition, for a pin pre-design study, a total helium release is still considered to determine changes in gas composition and pressure of the plenum. This assumption may seem inconsistent with the previous one adopted for helium retention evaluation, but the reason is that the combination of these two choices leads to maximize both the gaseous swelling induced by helium retention, and potentially the resulting fuel-cladding mechanical interaction, and also the plenum gas pressure evolution. In other words, the principle consists in maximizing the mechanical loadings on cladding, in an “envelope” pre-design approach.

II.1.2. Conductivity law

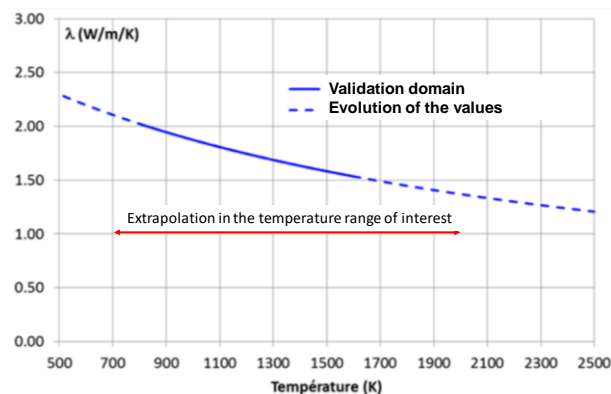
A specific thermal conductivity law was developed and implemented in GERMINAL for (U,Am)O₂ fuels. It consists of a law determined for a (U,Am)O_{2-x} mixed oxide sub-stoichiometric fuel sample (CP-ESFR4 sample [17]) with the following composition: U/M=85%, Am/M=15%, 1.93<2-x<1.95.

The variation of the conductivity of this fuel sample as a function of temperature is described by the following correlation:

$$\lambda_0(T) = 1 / (3.369 \cdot 10^{-1} + 1.966 \cdot 10^{-4} \times T)$$

This correlation is valid (i.e. deduced from thermal diffusivity experimental measurements) in the temperature range 800 K ≤ T ≤ 1600 K. However, as Figure 1 shows, given the slow evolution of the values, extrapolation of the correlation outside the validity range is reasonably acceptable between [700K-2000K], temperature range of interest for the present study.

Figure 1: Thermal conductivity of a (U,Am)O₂ sample as a function of temperature



II.1.3. Fuel restructuring and actinides redistribution

Fuel restructuring models (migration of porosities and formation of the central hole) and redistribution of actinides, used for standard (U,Pu)O₂ fuels, are disabled in the AmBB calculation scheme by appropriate parameters assignments. Actually, these models (determining oxygen potential, chemical potential of oxides in vapour phase, partial pressures of these oxides and associated fluxes in the pellet) are adapted to (U,Pu)O₂. As the different oxide species and thermodynamic balances are material-specific, these models are therefore not applicable to (U,Am)O₂ fuel, given the differences in material composition, the high amount of minor actinides and for which other species and balances should be considered.

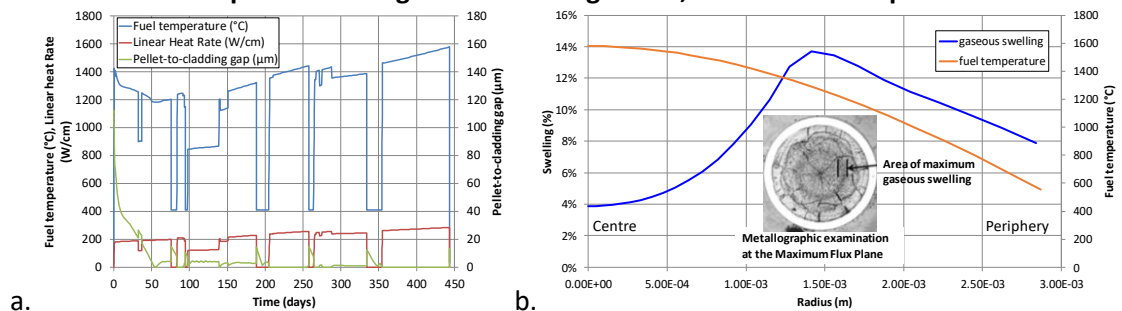
In addition, AmBB fuels contain throughout their stay in the core a low proportion of fissile isotopes, thus the operating thermal regime will be significantly lower than that of a fissile element. Therefore, it is reasonable to consider that the operating regime of AmBB fuels is significantly lower than the thermal activation thresholds of the transport/diffusion mechanisms leading to the migration of porosities and species. In other words, these mechanisms of fuel restructuring and actinides redistribution will not activate in AmBB fuels.

II.1.4. Verification

The verification of the AmBB calculation scheme is based on the simulation of the irradiation in PHENIX of SUPERFACT pin N°6 [6], [7]. This pin is constituted of (U,Am,Np)O_{2-x} fuel with (Am/M) = 19%, (Np/M) = 21% et 2-x = 1.926. The objective is to verify the predictions of the new gaseous swelling model. Figure 2 shows the calculated radial profile of gaseous swelling in the pellet at the maximum flux plane at end of irradiation (EOI), and the corresponding macrograph of the pellet.

Figure 2: Gaseous swelling of SUPERFACT pin N°6:

a. Linear Heat Rate, fuel temperature and pellet-to-cladding gap as a function of time;
b. Fuel temperature and gaseous swelling at EOI, as a function of pellet radius



The prediction for the maximum gaseous swelling is very close to 15%, considered as "envelope" value in order to increase the pellet-cladding mechanical interaction resulting from the closure of the gap caused by the swelling of the pellet. It occurs at a pellet radius of 1.4 mm ($\sim 0.5 r/r_0$). This is consistent with the fact that there is an intermediate area of the pellet with an optimum of gas retention and temperature leading to maximum swelling: the swelling decreases

- in the centre of the pellet, due to gas release which is predominant in high temperature regions,
- and at the periphery of the pellet, due to a lower volume fraction of precipitated gases at lower temperature.

The predictions of the gaseous swelling model are therefore physically consistent, and its amplitude in accordance with the targeted envelope value.

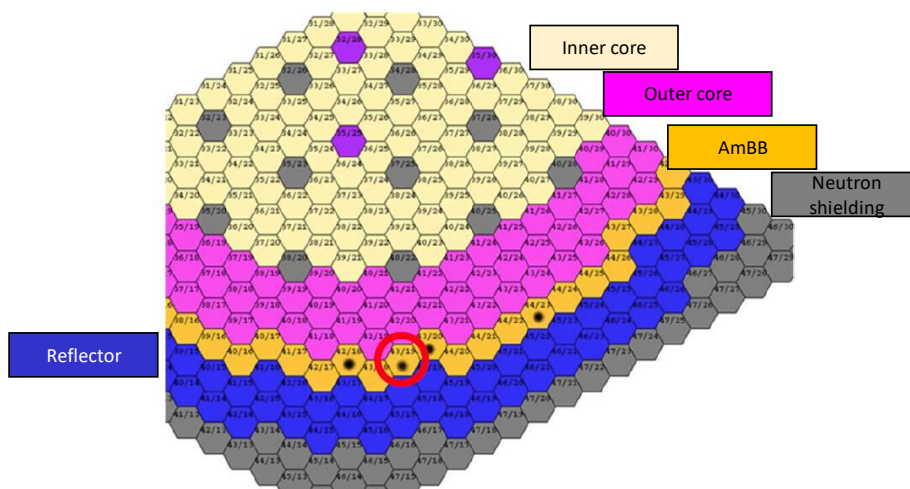
II.2. Simulation parameters of the AmBB pin reference case

The scenario, neutronics characteristics, sub-assembly / pin geometry and materials detailed in the following paragraphs constitute the reference case for the AmBB pin pre-design study.

II.2.1. Scenario and neutronics characteristics

Americium transmutation is supposed to take place in a 3600 MWth system core. A schematic diagram of a typical ASTRID-type core considered for this study is represented on Figure 3. This study concerns the hottest AmBB sub-assembly : red circle on Figure 3.

Figure 3: Position of the dimensioning sub-assembly in the core



The selected scenario is the following:

- irradiation duration: 1940 EFPD (5 cycles of 388 EFPD),
- no 180 degree rotation of the sub-assembly between cycles,
- inter-cycles duration not taken into account in the calculations,
- cooling down (zero power and 25 °C temperature) at the end of irradiation.

The fuel neutronics characteristics are:

- americium in a uranium oxide matrix : $(U,Am)O_{2-x}$,
- fuel fraction $(U,Am)O_{2-x}$ in the sub-assembly = 41.6%,
- chemical composition: $Am/(U + Am) = 10\%$,
- americium isotopic composition: 80% of ^{241}Am and 20% of ^{243}Am ,
- cross sections, dose, linear power evaluated by neutronic calculations and provided on input to GERMINAL V2.2

II.2.1. Sub-assembly / pin geometry and materials

The geometrical data of the sub-assembly and pin considered for the study are summarized in Table 1.

Table 1 : Geometrical data of the AmBB pin and sub-assembly

Data	Value	Data	Value
Pellet diameter	$8.45 \cdot 10^{-3}$ m	Internal width of hexagonal wrapper tube	0.2083 m
Initial pellet-cladding gap	125 μ m	External width of hexagonal wrapper tube	0.1993 m
Cladding internal diameter	$8.7 \cdot 10^{-3}$ m	Inter-assembly gap	$4.5 \cdot 10^{-3}$ m
Cladding external diameter	$9.7 \cdot 10^{-3}$ m	Neutronic mesh	$2.25 \cdot 10^{-3}$ m
Spacer wire diameter	$1.0 \cdot 10^{-3}$ m	Number of rows	8
Spacer wire step	0.225 m	Inferior fertile zone height	0.3 m
Pin mesh	$13.18 \cdot 10^{-3}$ m	Na section of the bundle	$1.82 \cdot 10^{-2}$ m ²
Inferior plenum height	0.830 m	Wet perimeter of the bundle	7.99 m
Superior plenum height	0.071 m	Hydraulic diameter of the bundle	$9.1 \cdot 10^{-3}$ m

The cladding is in austenitic steel 15-15 Titanium grade D4, precursor of AIM1 [18].

The (U,Am)O_{2-x} fuel has the following characteristics:

- density of the pellet compared to its theoretical density: 88%,
- porosity: 12%, essentially open porosity,
- O/M ratio: 1.995.

III. Results of pre-design of an Am-BB pin

The evolution of parameters with irradiation, fuel and cladding behavior are discussed in the first paragraph of this section, for the AmBB pin reference case (without uncertainties). Then, the pre-design criteria and corresponding calculated pre-design results are detailed. Finally, in the last paragraph of this section, the uncertainties on the data related to the geometry and the behaviour of the pellet and cladding are taken into account and the results are presented.

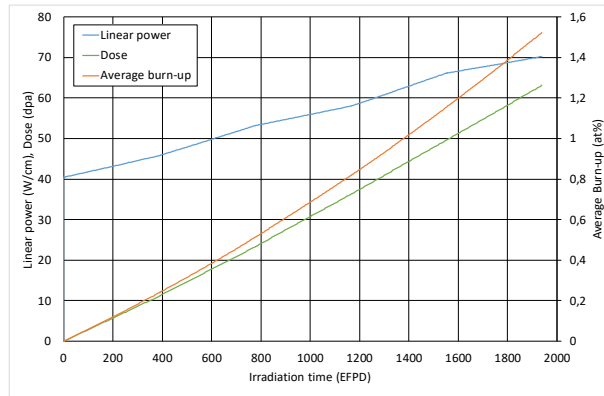
III.1. Evolution of parameters with irradiation, fuel and cladding behavior

III.1.1. Linear power, dose and burn-up

The linear power, burn-up and dose evolution with irradiation are represented on Figure 4. The maximum Linear Heat Rate (LHR) at the Maximum Flux Plane (MFP) is 70.2 W/cm at the End Of Irradiation (EOI). This value is low compared to that of a standard fuel (> 400 W/cm). The consequence of this low value is that the fuel remains “cold”, which limits both the internal pressure in the fuel pin and the release of helium and gaseous fission products. This could lead to an increased swelling of the pellet due to this high gas retention (see section III.1.3).

At the end of irradiation (1940 EFPD), the maximum dose is 63.1 dpa and the averaged burn-up over the height of the fissile zone is only 1.52 at% (14,6 GWj/t_{HM}). This low value, related to the assumptions on irradiation duration and Am content, will lead to a low solid swelling of the fuel. Its averaged value is estimated to be about 1%, but is quite dependent on temperature. The maximum burn-up at MFP at EOI is 2.4 at%, corresponding to 45 μm increase of pellet diameter.

Figure 4: Linear power, dose and burn-up as a function of irradiation time

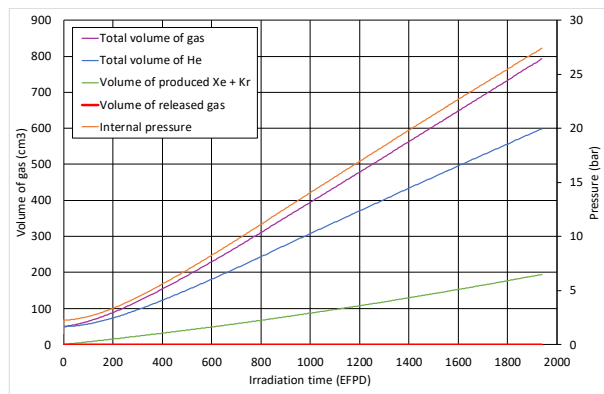


III.1.2. Helium and fission gas production and release, internal pressure

Figure 5 shows helium and Fission Gases (FG) Xe and Kr production and release, as well as the internal pressure of the pin as a function of irradiation time. The volume of gas formed at EOI (at hot state) is 194.5 cm³ for FG (green curve) and 598 cm³ for helium (blue curve). As specified above, as the fuel remains cold, the FG release is negligible (red curve), and the consequence is an almost total gaseous retention with a potential gaseous swelling of the pellets.

The maximum value of the internal pressure in the fuel pin is 27.4 bar (orange curve) with the assumption of total helium release. With this conservative assumption, this value is compatible with quite high margins with regards to the RAMSES thermo-mechanical criteria (see III.2.2).

Figure 5: Helium and fission gas production and release, internal pressure

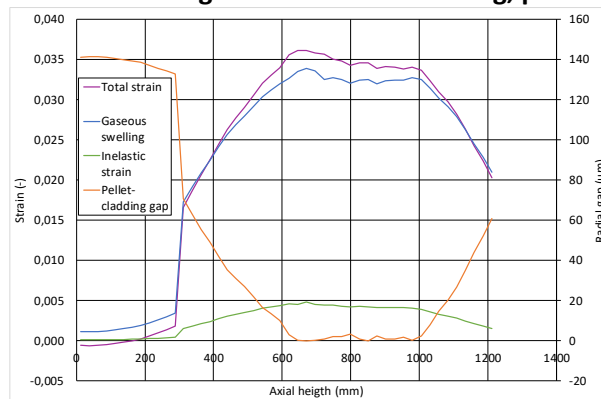


III.1.3. Strain and swelling of the fuel and cladding, pellet-cladding gap

Figure 6 shows the main mechanisms contributing to the fuel total deformation (purple curve): gaseous swelling (blue curve) and inelastic deformation (green curve), as a function of the pin height. The closure of the pellet-to-cladding gap (orange curve) can be seen between 622 mm and 958 mm (0.52 and 0.8 h/h₀), which is consistent with Figure 7 where the fuel surface temperature matches the cladding internal surface temperature (see III.2.1).

In all the following graphs representing the evolution of the parameters as a function of the pin axial height, 0 mm represents the bottom and 1200 mm the top of the pin.

Figure 6. Strain and swelling of the fuel and cladding, pellet-cladding gap



III.2. Design criteria

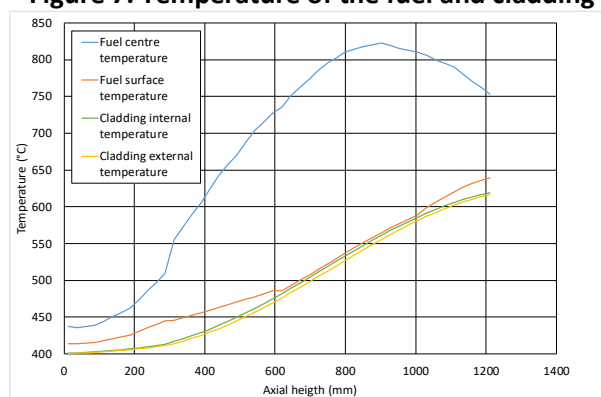
III.2.1. Thermal design criteria and temperature of the fuel and cladding

In this study, the thermal criterion for the fuel pellet consists in the verification that the margin to melting is higher than 300 °C; this margin is usually considered for (U,Pu)O₂ fuel [19].

Another thermal criterion is the Nominal Cladding Temperature (NCT), which is the maximal temperature of the hottest cladding of the sub-assembly. It is imposed to be NCT = 620°C to avoid cladding thermal-activated creep and to preserve margin to melting in fuel. The respect of this criterion is guaranteed by the sub-assembly cooling (see III.2.1).

We can see on Figure 7 that at EOI, the fuel centre temperature does not exceed 825 °C. The margin to melting is considerable (~ 2000 °C). Another consequence of this low thermal regime is that no central hole could form in the pellet. On either side of the MFP, the fuel surface temperature matches the cladding internal surface temperature, showing a closure of the pellet-to-cladding gap, probably leading to fuel cladding mechanical interaction (see also Figure 6).

Figure 7. Temperature of the fuel and cladding



III.2.2. Thermo-mechanical criteria (RAMSES criteria) and thermo-mechanical behaviour of the cladding

Three criteria define the margins to failure of the cladding. They are the so-called RAMSES criteria.

- R_1 criterion relates to the primary membrane stress (P_m); it has to be higher than 1 at any time during irradiation in order to validate the design of the cladding (for this criterion):

$$R_1 = \frac{\min(\frac{R_m}{3}, 0.8R_{p0.2})}{P_m} > 1$$

with R_m : tensile strength, $R_{p0.2}$: yield stress for 0.2% plastic deformation and P_m : general primary membrane stress

- R_2 criterion shows, in addition to the primary membrane stress, the general flexion stress, $|P_1+P_b|$; it has to be higher than one at any time during irradiation in order to avoid reaching the limits of the cladding material:

$$R_2 = \frac{\min(K\frac{R_m}{3}, 0.8R_{p0.2})}{P_1 + P_b} > 1$$

with P_1+P_b : general primary bending stress and $K = 1.5$ for the cladding in nominal operating conditions

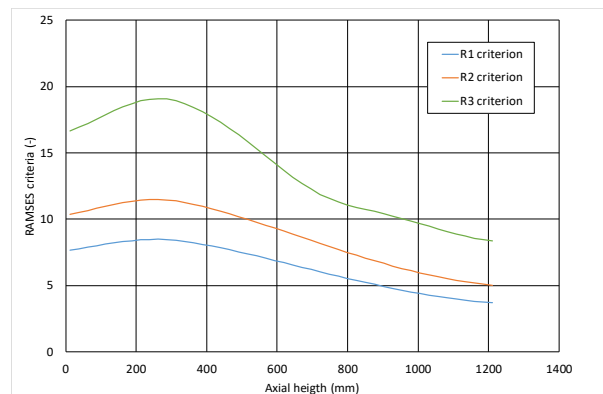
- R_3 criterion compares the total local Von Mises stress (σ) with the local material properties ($0.8R_{p0.2}$); it has to be higher than one to avoid reaching the limits of the cladding material:

$$R_3 = \frac{0.8R_{p0.2}}{\sigma} > 1$$

- An additional criterion for creep usage fraction, U , has been defined; it has to be less than 1 at any time during irradiation in order to avoid reaching the creep limits of the cladding material.

Figure 8 shows the evolution of R_1 , R_2 and R_3 criteria along the pin, for the studied AmBB pin reference case. The three criteria are minimum at EOI, at the top of the pin: $R_{1min} = 3.7$, $R_{2min} = 5$ and $R_{3min} = 8.3$. The three RAMSES criteria are widely respected. In addition, the U criterion for creep usage fraction is $U \sim 7.3 \cdot 10^{-9} \ll 1$. The criterion is respected and the margin is huge.

Figure 8. RAMSES criteria along the pin at EOI



III.2.3. Thermal-hydraulic pre-design criteria and results

The AmBB sub-assembly must be coolable. Since the nominal cladding temperature is 620°C in the GERMINALV2.2 calculations, the thermal-hydraulics calculations first determine the necessary flow rate to guarantee this temperature when the highest power of the sub-assembly is reached. Once the flow rate known, the pressure loss in the pin bundle of the AmBB sub-assembly is deduced. The calculated values are then compared with the flow rate and pressure loss of a typical ASTRID bundle in order to check that:

- the cooling flow rate is compatible with the flow rate range of SFR sub-assemblies, among which ASTRID ones (~4 kg/s to ~40 kg/s),
- the bundle pressure loss is acceptable and compatible with the loss of total core load (i.e. lower and limited to a maximum between ~1.5 and ~1.8 bar, bundles with lower pressure may require the use of a diaphragm to reach these values).

For these thermal-hydraulic calculations, the evaluation of cooling flow rate and pressure loss of the bundle is performed with the assumption of independent sub-channels. This approach is conservative since it maximizes the flow rate and pressure loss. The sub-assembly is supposed to be adiabatic.

The following results are obtained:

- the dimensioning time is EOI,
- the required flow rate to cool the bundle is ~ 7 kg/s,
- the pressure loss related to the bundle is ~ 0.013 bar,
- at the beginning of the cycle the average bundle temperature is ~ 74 °C, and the NCT is 534 °C,
- at the end of the cycle the average bundle temperature is ~ 123 °C, and the NCT is 620°C.

In conclusion, the required flow rate to cool the bundle is consistent with an ASTRID sub-assembly, but the bundle pressure loss is much lower than that of the ASTRID core. Such a sub-assembly will require the use of a diaphragm to achieve a higher pressure loss.

III.3. Influence of uncertainties

Uncertainties are not taken into account in the results of the previous section (reference case). Uncertainties to be considered should be high due to high uncertainties on:

- gap size (linked to gaseous swelling),
- gaseous release,
- linear heat rate (due to neutron flux gradient),
- fuel composition and thermal conductivity.

The influence of some of them are taken into account in the present section. The objective of this study is not to assess the influence of all uncertainties since they are not yet all known at this stage. The conservatisms taken into account in this study are the following:

- minimum cladding thickness = 0.46 mm (nominal value used for fuel sub-assemblies = 0.5 mm),
- diminution of the expansion volume height (EVH) in the ratio as follows: x0.99 for the lower EVH and x0.96 for the upper EVH,
- complete release of helium and fission gases,

- maximization of internal and external corrosion speed (maximum standard laws in GERMINAL V2),
- maximum cladding temperature = 630 °C.

The results show that these different conservatisms don't modify the conclusions obtained for the reference case. The thermal and thermo-mechanical (RAMSES criteria) pre-design margins are still high, as shown in Table 2, which summarises the results obtained for the reference case and for the case with uncertainties.

Table 2: Comparison between results obtained for the reference case and for the case with uncertainties

Data	Reference case	Case with uncertainties
Initial pellet diameter	8.45 .10 ⁻³ m	8.45 .10 ⁻³ m
He volume produced	598 cm ³	598 cm ³
FG volume produced	194 cm ³	194 cm ³
Pellet-cladding gap at EOI	Gap closure	No gap closure
Pressure	27.4 bars	36.3 bars
R1 criterion	3.7	3.3
R2 criterion	5.0	4.5
R3 criterion	8.3	7.5
Creep usage rate	7.3 .10 ⁻⁹	2.1 .10 ⁻⁸

IV. Conclusion

This paper presents the results of a pre-design study of an AmBB pin performed with the GERMINAL V2.2 fuel performance code of the PLEIADES simulation platform.

A calculation scheme dedicated to the AmBB pin was developed and implemented with the following specificities:

- gaseous swelling model correlated to fission gas retention and helium production,
- internal pressure calculation assuming total helium release,
- thermal conductivity law specific to (U,Am)O₂ fuel,
- no activation of the standard (U,Pu)O₂ fuel restructuring (migration of porosities and formation of the central hole) and redistribution of the actinides.

The data used in this study are the following:

- geometry of the fuel sub-assembly of ASTRID-type outer core,
- 10% americium loaded in a UO_{2-x} matrix,
- irradiation time: 1940 EFPD in five operating cycles of 388 EFPD,
- no 180 degree rotation of the sub-assembly at mid-life.

The results of this study show that all the pre-design criteria are respected and the thermal and thermo-mechanical margins are high, even when taking into account the uncertainties. A first thermal-hydraulic pre-design of the reference case, performed with the assumption of independent sub-channels (envelope approach) shows that the required flow rate and pressure loss remain compatible with that of SFR, including ASTRID.

These conclusions open the path, from the AmBB pin behaviour point of view, to improvement possibilities. The AmBB pin pre-design shows high margins, suggesting that increasing both the Americium content and the irradiation time could be possible (without considering the constraints related to the fuel cycle). Considering the fuel thermal margins and the cladding thermo-mechanical margins (low internal pressure), a promising improvement could be to increase the pin diameter. The consequence would be an increase of the fuel proportion in the sub-assembly, leading to an increase of the linear power at the end of irradiation. Thus, the fuel temperature would be higher, which would enhance gas release and reduce swelling and risk of pellet-cladding mechanical interaction. Another potential advantage would be a lower number of pins to fabricate, but with a different geometry than standard driver fuel pin. In parallel, R&D efforts on fuel fabrication have been pursued to increase open, interconnected fuel porosity in order to favour gas release.

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