Design of a Nitride-fueled Lead Fast Reactor for Minor Actinides Transmutation

P. Larroche[1], S. Bortot[1], J. Wallenius[1,2], I. Mickus[2]

[1]Kungliga Tekniska Högskolan (KTH), Stockholm, Sweden

[2]LeadCold Reactors, Stockholm, Sweden

E-mail contact of main author: larroche @kth.se

Abstract. The core design of a 600 MW$_e$ lead-cooled, nitride-fueled fast reactor aimed at transmuting MAs is presented. The following major goals are pursued: (*i*) obtaining a unitary conversion ratio; (*ii*) achieving a 6 kg·TW$_e$h^{-1} specific Am consumption after 6 years of cooling in a homogeneous transmutation scenario, while (*iii*) respecting the fuel cycle constraints for fuel maximum thermal load of 7.5 kW per assembly after 5 years cooling and of 3 kW per fresh assembly. The core neutronics characterization is performed with the Monte Carlo code Serpent. Complementary design-oriented transient analyses are finally carried out by means of BELLA, a dynamics code jointly developed by KTH and LeadCold for the safety analysis of Generation-IV innovative lead fast reactor systems. The core transient behavior following postulated accident initiators is simulated and reference safety criteria, such as margins against cladding failure, fuel melting and nitride dissociation, are consequently assessed.

Key Words: Generation-IV Lead Fast Reactor (LFR), MA transmutation, core design, safety performance.

1. Introduction

Research and development of fast reactors have is being carried out in several European nations with mixed oxide fuel and sodium coolant as the reference materials. In addition, the transmutation of Minor Actinides (MAs) in fast reactors is being investigated extensively with the aim to reduce the environmental burden of long-lived radioisotopes.

Nitride fuels and lead as coolant represent possible alternatives for reactor design, which can bring substantial advantages when compared to oxides and sodium, respectively. Among the main reasons for that, it is noted that nitride fuels exhibit a higher thermal conductivity than that of oxides. Therefore, the larger margin provided against fuel failure enables to survive higher magnitude overpower transients. Since americium brings detrimental consequences on safety parameters, such enlarged margins to fuel failure allow for larger amounts of MAs loading, as compared to the affordable quantities for oxide-fueled cores guaranteeing analogous safety features.

Concerning spectrum-related benefits, higher neutron energies brought by the combined use of nitride fuel and lead coolant favor better MA burning performances, which compensate for the typically lower Doppler constants.

As far as global safety aspects are concerned, one important flaw of Sodium-cooled Fast Reactors (SFRs) is represented by the violent exothermal reactions between sodium and water or air, which impose the use of an intermediate loop, with consequent penalty on the economic side. On the other hand, lead does not react with water or air, has a considerably

higher boiling point than sodium, and its density changes largely with temperature. These properties allow for a simplified design of the primary loop, and the opportunity to rely on natural circulation cooling in case of accidental conditions such as Loss Of Flow (LOF). Such advantages justify the increasing interest for Lead-cooled Fast Reactors (LFRs), despite their major drawback: the corrosive/erosive nature of lead on conventional steels, which imposes tight constraints on both the selection of candidate materials and temperatures [1].

In this paper, the work performed to design a LFR aimed at transuranic (TRU) transmutation and electricity production while meeting the target safety performances is presented.

2. Core Design

In this section, the main design goals and priorities are discussed along with the interconnections and mutual implications among parameters. It is pointed out that the presented core configuration does not include detailed engineering solutions, as it consists in a conceptual design based on stationary and transient neutron physics and thermal-hydraulics.

2.1. Goals and Constraints

The primary design goal consists in designing a commercial reactor that is able to use previous generation reactor waste as fuel and to burn a sufficient amount of americium. The aimed core thermal power, americium burning rate and Conversion Ratio (CR) are specified in in Table I.

TABLE I: MAIN CORE DESIGN GOALS.

Parameter	Value
Thermal power	1500 MW
Specific americium consumption after 6 years of cooling	$6 \ kg \cdot TW_e h^{-1}$
Peak burn-up	$100 \ GWd \cdot t^{-1}$
Conversion Ratio	1 ± 0.05
Minimum cycle length	365 days
Minimum residence time	4 cycles
Maximum assembly decay heat after 5 years of cooling	7.5 kW
Maximum assembly fresh fuel thermal load	3 kW

Technological constraints and expected safety performances are to be added to these goals in order to fully fulfill the design specifications.

Among the former, specified in Table II, the most stringent requirements concern the 15-15Ti cladding[1] maximum temperature, which needs to be maintained under 550 °C in operating conditions to prevent corrosion. In addition, the coolant velocity needs to be limited to $2 \ m \cdot s^{-1}$ in order to avoid cladding erosion [2]. Moreover, the coolant temperature range is set so as to allow sufficient margins above its freezing point (327 °C) and below the cladding maximum temperature. The fuel temperature is to be kept under the nitride dissociation temperature [3].

---

[1] The choice of 15-15Ti as the cladding material is due to its capability to withstand high fluences up to 100 dpa [4], as demonstrated in Phénix; the choice of austenitic steels is instead driven by their high tolerance against thermal creep [5].

TABLE II: MATERIAL-RELATED TECHNOLOGICAL CONSTRAINTS.

Parameter	Value
Maximum fuel temperature in unprotected accident	2400 °C
Maximum clad temperature in normal operation	550 °C
Maximum clad damages	100 dpa
Nominal coolant inlet temperature	400 °C
Nominal coolant outlet temperature	500 °C
Maximum coolant velocity	2 m·s^{-1}
Maximum clad temperature during unprotected accident	750 °C

2.2. Design Approach and Methodology

In order to accomplish the primary transmutation goal, the core design methodology must necessarily start with the determination of the fuel composition allowing the desired CR in a first-guess energy spectrum. Subsequently, among the possible fuel compositions, the option with the lowest americium concentration providing a 6 kg·TWh^{-1} americium consumption is chosen, so as to be likely complying with the safety requirements by limiting the detrimental effects of MAs on both kinetic parameters and reactivity coefficients.

First-guessed pin and sub-assembly specifications with the calculated fuel concentration are input in the Monte-Carlo code Serpent [6] to assess the corresponding energy spectrum and neutron flux. This iterative process finally leads to the fuel composition and sub-assembly specifications used to build the full core configuration. The latter is optimized by foreseeing different radial zones characterized by the same average fuel composition, with the dual objective to reach the lowest reactivity swing and the lowest radial power distribution factor.

Once completed the core static neutronics calculations, including kinetic parameters and reactivity coefficients, the core dynamics is investigated in order to assess the system behavior in case of postulated accidents, and the design is consequently refined and, eventually, finalized.

2.3. Fuel Composition

As the priority was set on the transmutation performance, the first parameter to determine is the fuel average composition. The fuel consists in a mixture of plutonium, americium and depleted uranium. The selected actinide isotopic vectors are consistent with the standard French Light Water Reactors (LWRs) spent fuel composition. Given the aimed quantity of americium to be transmuted and once set the desired conversion ratio, a Bateman system relative to the fuel vector is solved in order to find its optimal composition.

The Bateman system can be written as a matrix equation:

$$\frac{d\vec{N}(t)}{dt} = A \times \vec{N}(t) \tag{1}$$

where $\vec{N}(t)$ is the isotopic mass fraction at time t, and A the transmutation matrix, depending on the neutron flux, initially assumed to be constant. The solution of this system is the vector:

$$\vec{N}(t) = e^{At}\vec{N_0} \tag{2}$$

where $\vec{N_0}$ is the isotopic mass fraction at Beginning of Life (BoL). Therefore, by applying constraints on $\vec{N}(t)$ at End of Life (EoL), and on $\vec{N_0}$ at BoL, only the fuel composition

leading to the specified conversion ratio is kept. More specifically, the constraints were applied on the mass fraction of ^{241}Am, ^{243}Am, and fissile isotopes (in particular ^{239}Pu and ^{241}Pu) at BoL and EoL.

Nineteen isotopes were considered in the model, with the neutron flux being first set to a realistic value of 10^{15} n·s^{-1}·cm^{-2}

After solving the Bateman system and determining the first-iteration fuel composition, the latter was used to perform a cell calculation with SERPENT on a single fuel pin. The resulting average fuel composition at BoL is presented in Table III.

TABLE III: AVERAGE FUEL COMPOSITION[2].

Molecule	Mass fraction
UN	77.5 %
PuN	17.6 %
AmN	4.9 %

With this composition, a 0.98 conversion ratio is achieved, and the target specific americium consumption is very well met.

2.4. Pin and Sub-channel Design

The pin and sub-channel specifications have to be designed in order for the fuel, cladding and coolant to be at their nominal temperature during normal operation (see Table I).

Starting from the fuel rod geometrical specifications of the Advanced Lead-cooled Fast Reactor European Demonstrator (ALFRED) core configuration [7] as the first-guess parameters, the cladding outer diameter is established and iterations are performed to consequently determine the fuel pellet diameter, gap and cladding thickness, pin-pitch and core average linear power, so as to meet the desired cladding temperature constraint[3].

The gap thickness is chosen to cope with a swelling consistent with a 9 % burn-up - which is confirmed to be a reasonable assumption by depletion calculations - assuming that no interactions occur between pellet and cladding[4]. In addition, the pellet is hollowed in order to guarantee additional margins to swelling-related issues.

The pin-pitch and fuel pellet size are determined with the twofold purpose of reaching a fuel volume fraction consenting core criticality, and of limiting the coolant velocity to 2 m·s^{-1}. Following a meticulous neutronic and thermal-hydraulic (T/H) optimization, the sub-channel geometry parameters are eventually determined (see Table IV).

The core active height is fixed to 100 cm, the linear power being expected to result between 320 and 340 W·cm^{-1}.

The gas plenum is designed so that the stress on the cladding due to the internal pressure induced by fission gas release remains under 100 MPa [4]. In this model, only the Helium produced by alpha-decay is taken into account, along with the assumption than the latter is

---

[2] Average fuel density of 1282 kg·m^{-3} with a smear density of 90 %.

[3] One can notice that with the use of nitrides, contrary to oxides, the fuel peak temperature is not critical for the pin design.

[4] The model used to predict the swelling is the empirical correlation proposed by Ross et al. [8].

entirely released into the gas plenum. As a result, a 100 cm gas plenum is required, evenly located above and below the active zone.

TABLE IV: SUB-CHANNEL HOT-STATE GEOMETRICAL PARAMETERS.

Parameter	Value
Pellet hole diameter	2.0 mm
Pellet outer diameter	9.3 mm
Inner cladding diameter	9.5 mm
Outer cladding diameter	10.5 mm
Pin-pitch	16.6 mm
Hydraulic diameter	15.9 mm
Linear power	326 $W \cdot cm^{-1}$
Lead velocity	1.56 $m \cdot s^{-1}$

A preliminary verification of the average-pin peak cladding temperature is performed by adopting a simplified lumped-parameter core sub-channel T/H model: the outer cladding temperature is predicted to be equal to 511 °C in nominal conditions, which guarantees a reasonable margin against the limit discussed above. It is noted, however, that such a model cannot provide information about the hottest pin in the assembly.

2.5. Controls Rods

The criteria adopted to design control and safety assemblies[5] are the following:

- absorbers must compensate the burn-up reactivity swing;
- absorbers must compensate for the provided excess reactivity in cold state allowing to reach criticality in hot state;
- absorbers must have a sufficient worth to SCRAM the reactor in case of accident.

Moreover, a further requirement to limit the worth of the most critical absorber assembly to less than 1 $ is imposed, so as to prevent prompt-criticality in case of inadvertent control rod ejection.

The cold-to-hot state worth is predicted to be of the order of 2 $, and the reactivity swing turns out to be less than 2 $, as detailed in Section 3. Considering, additionally, a best-practice recommendation to provide a 10 $ reactivity margin for SCRAM, the total absorber worth must exceed 15 $.

The absorber material chosen for control elements is B_4C with 2.2 $g \cdot cm^{-3}$ density[6], and 48 % enrichment in ^{10}B.

In order to evaluate the actual absorber worth, a homogenized control element model with a 25 % B_4C volume fraction is implemented in Serpent. For a total of 32 rods, the absorber reactivity worth is displayed as a function of the insertion length *(see FIG.1.)*. The worth of one average control rod is 0.88 $ would correspond to a total worth of 28 $, if the shadow

---

[5] No distinction between the two is made in this work, resulting in the same design for both control and safety assemblies.

[6] The average density of the absorbers being lower than that of the coolant, control and safety rods are located under the active core; safety rods are passively inserted by buoyancy.

effect was neglected. Accounting also for the reactivity worth overestimation caused by homogenization (which reduces self-shielding), the resulting global absorber worth is predicted to be of the order of at least 16 \$, which leaves some margin for a further optimization and engineering of the control system.

At BoL, the core is supercritical when all the control rods are extracted; consequently, criticality is reached by inserting the absorbers bank by 10 cm.
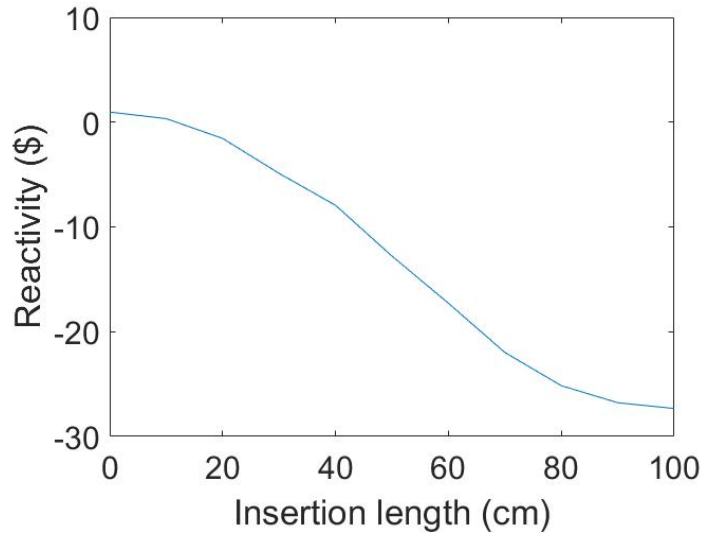

FIG. 1. Control rod worth as a function of the insertion length.

2.6. Core Geometry and Arrangement

Once defined the sub-assembly specifications and control rod parameters, the full core geometrical configuration is developed so as to reach the best power flattening while aiming for the target transmutation performance and power level. As far as the radial power flattening is concerned, the initial intent to limit the power distribution factor below 1.1 is slightly released, since the value of 1.11 characterizing the selected core configuration results largely acceptable.

The final core is divided into three radial zones, as described in Table V and depicted in *FIG. 2*. Two rings of dummy assemblies surround the active core in order to limit neutron leakage.

Control rods are symmetrically distributed across the core; three additional absorber assemblies are located in the last ring of the inner zone and are inserted since BoL, their primary function being the reactivity margin compensation in cold state.

TABLE V: CORE ZONES SPECIFICATIONS.

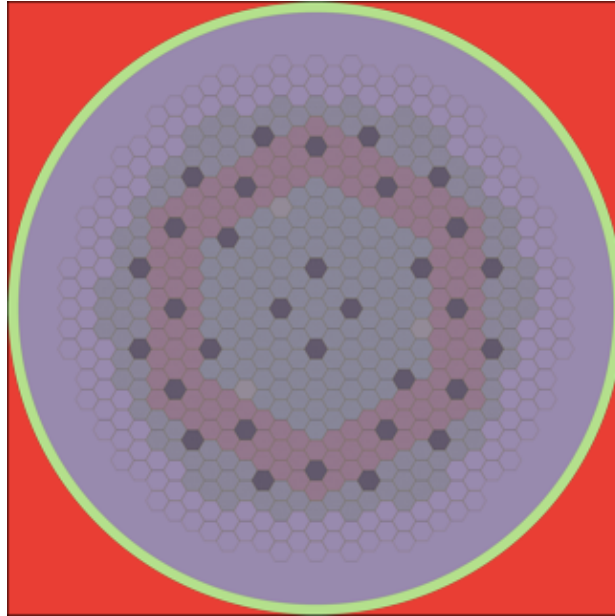| Parameter	Inner zone	Middle zone	Outer zone
UN mass fraction	80.2 %	79 %	73 %
PuN mass fraction	15.8 %	17 %	20 %
AmN mass fraction	4 %	4 %	7 %
Number of assemblies	116	132	114

FIG. 2. Final core layout.

3. Neutronic Characterization

The burn-up targeted *a priori* is 100 MW·d·kg^{-1}. However, due to swelling-related issues, this aim is necessarily lowered down to 90 MW·d·kg^{-1}; with a maximum linear power of 326 W·cm^{-1}, such a burn-up is achieved after 6 years of irradiation.

The resulting reactivity swing is depicted in *FIG. 3*. As mentioned above, it can be observed that, in order to achieve a burn-up of 90 MW·d·kg^{-1}, it is necessary to provide a 2 \$ excess reactivity at BoL. According to the general design requirements (see Table II), one fourth of the core is to be refueled every 1.5 years.

The transmutation of americium after the full irradiation of 90 MW·d·kg^{-1} results 6.45 kg·TW^{-1}·h^{-1} (*FIG. 4*). The corresponding decay heat per assembly after 6 years of cooling is 2.3 kW, calculated only by accounting for the contribution of MAs decays[7].
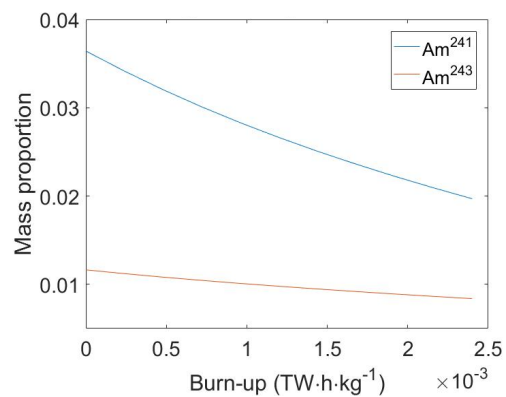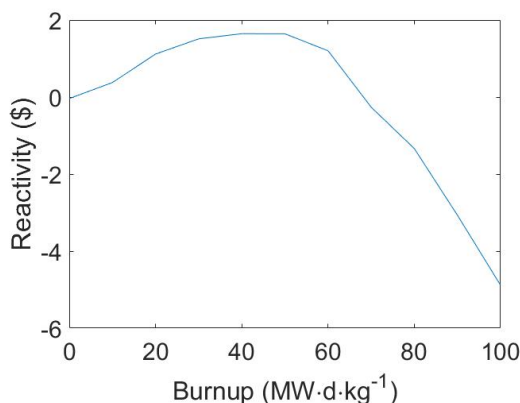

FIG. 3. Reactivity swing as a function of burn-up.   *FIG. 4. Americium burning performances.*

Core reactivity feedbacks and kinetic parameters calculated at BoC are presented in Table VI. The values of the radial expansion coefficient are computed considering the diagrid made of

---

[7] The fresh fuel thermal load is evaluated to 1 kW per assembly.

austenitic steel or ferritic-martensitic steel; the former is used in the subsequent transient analysis.

TABLE VI: CORE SAFETY PARAMETERS AT BoL.

Parameter	Value
Doppler coefficient	-0.33 pcm·K^{-1}
Core coolant density coefficient	+0.54 pcm·K^{-1}
Axial expansion coefficient	-0.15 pcm·K^{-1}
Radial expansion coefficient (SS steel/FM steel diagrid)	-0.90/-0.67 pcm·K^{-1}
Effective delayed neutron fraction	330 pcm

4. Core Transient Analyses

A preliminary core transient analysis is performed by means of the zero-dimensional code BELLA [9] in order to finalize the static neutronics design by verifying if technological constraints are respected also beyond nominal conditions.

BELLA is a lumped parameter (0-D) system code under development by LeadCold for transient analyses of LFRs. The code solves coupled neutron kinetic and thermal-hydraulic equations, which allows investigating the time-dependent behavior of integral feedback effects and parameters (*e.g.* thermal power, temperatures, mass flow rates, etc.) important for system design and safety. The intended use of BELLA in the present work is motivated by the need to support a safety-informed core design.

The events simulated consist mainly in the result of (possibly multiple) faults combined with safety system failures, and are therefore qualified as unprotected transients.
More specifically, three reference scenarios are selected to assess the behavior of the core following postulated deviations from the nominal operating conditions:
  - Unprotected Transient Over Power (UTOP): the extraction of a control rod from the core is simulated by an insertion of 1 $ reactivity in 20 seconds;
  - Unprotected Loss Of Flow (ULOF): a malfunction in the pumping system is simulated by a total pump coast-down in 10 seconds, causing the core cooling to rely only on natural circulation cooling;
  - Unprotected Loss Of Heat Sink (ULOHS): a steam generator (SG) failure leading to the impossibility to exchange heat between primary and secondary system is simulated by hindering any heat removal in the SG within 10 seconds.
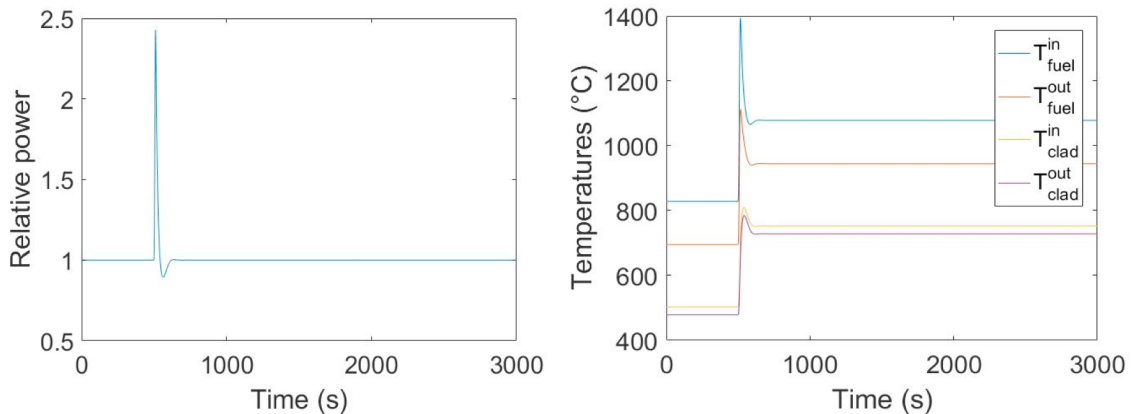

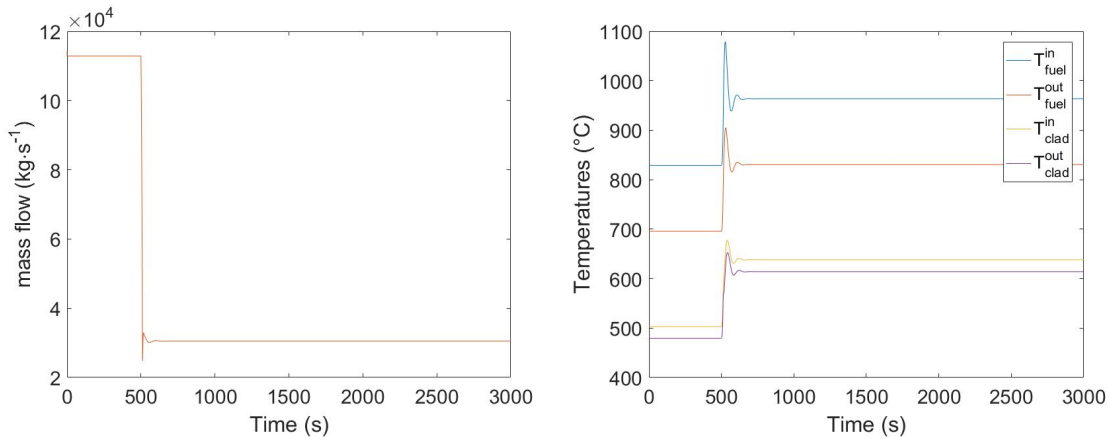*FIG. 5. Relative power and fuel rod temperatures time evolution (UTOP).*

FIG. 6. Core mass flow rate and fuel rod temperatures time evolution (ULOF).
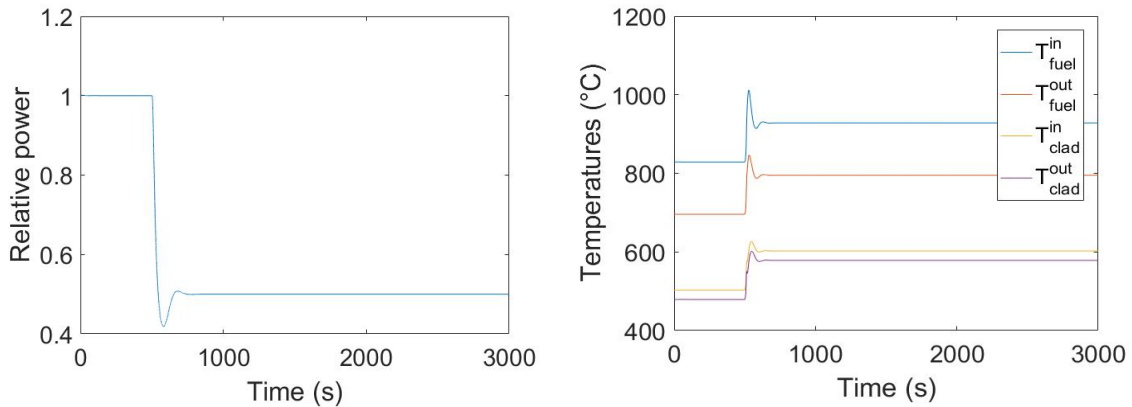

FIG. 7. Relative power and fuel rod temperatures time evolution (ULOHS).

The time-dependent behavior of the most relevant parameters following the postulated accident initiators is depicted in *FIGS. 5*, *6*, and *7*, respectively.

In case of UTOP, ULOF and ULOHS scenarios, significant margins to coolant boiling (1749 °C), nitride fuel dissociation (2400 °C), and cladding rapid creep failure (930 °C) are retained, favored by an overall negative power feedback coefficient. It is noted that natural circulation cooling allows the core to survive a ULOF accident, essentially contributing to the inherent core safety.

However, it is recalled that these results refer to the core average pin; consequently, more accurate (1-D) analyses are necessary in order to verify the conditions of the core hottest pin during transients, especially in the case of UTOP.

5. Conclusions

The conceptual core design of a 1500 MW_t LFR incorporating nitride fuel is undertaken with the major goals to obtain a unitary conversion ratio, a 6 kg $TW_e h^{-1}$ specific Am homogeneous consumption after 6 years of cooling, and a maximum thermal load of 7.5 kW per assembly after 5 years cooling and of 3 kW per fresh assembly.

Results show that the proposed core configuration satisfactorily meets the prescribed design goals as far as fuel-cycle- and power-related requirements are concerned, while respecting all technological limits with good margins.

The complementary analysis of the core transient behavior following postulated accident initiators confirms that the safety reference criteria are respected also when deviations from the nominal operating conditions occur, as cladding failure, fuel melting and nitride dissociation are prevented with fairly good margins. Moreover, inherent passive safety is guaranteed by the combination of the high fuel thermal conductivity, the overall negative power coefficient, and the proven great potential for natural circulation that lead coolant offers.

It is finally remarked that the cladding surface temperature appears to be the most critical parameter. Therefore, more accurate transient analyses are required in order to definitely assess the core safety performance and confirm its capability to survive severe accidents.

References

[1]   WEISENBURGER, A., et al., "Long term corrosion on T91 and AISI1 316 steel in flowing lead alloy and corrosion protection barrier development: experiment and models", Journal of Nuclear Materials, **415** (2008).

[2]   WEISENBURGER, A., et al., "Materials For ALFRED and ELFR – selection and challenges", Proc. FR13, Paris, France (2013).

[3]   CEA, "Bilan des recherches conduites sur la séparation-transmutation des éléments radioactifs à vie longue et sur le développement de réacteurs nucléaires de nouvelle génération, Tome 2", CEA, Paris, France (2012).

[4]   MAILLARD, A., et al., "Swelling and irradiation creep of neutron irradiated 316T1 and 15-15Ti steels", Proc. 16th Int. Symp. Effect of radiation on material, Denver, USA (1994).

[5]   WALLENIUS, J., "Transmutation of Nuclear Waste", LeadCold Editions, Stockholm, Sweden (2011).

[6]   http://montecarlo.vtt.fi/development.htm

[7]   GRASSO, G., et al., "The core design of ALFRED, a demonstrator for the European lead-cooled reactors", Nuclear Engineering and Design, **278** (2014).

[8]   ROSS, S.B., et al., "Uranium nitride fuel swelling correlation", Journal of Nuclear Materials, **170** (1990).

[9]   BORTOT, S., et al., "BELLA: a multi-point dynamic code for safety-informed design of fast reactors", Annals of Nuclear Energy, **85** (2015).