Fuel Performance Benchmark for the European Sodium Fast Reactor with Metallic Fuel

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ABSTRACT

Oxide fuel is the reference option for the European Sodium Fast Reactor (ESFR) original design as well as the one with the largest commercial experience world-wide. Conversely, the United States are focusing on metallic fuel for the first generation of commercial fast reactors, profiting from an extensive experience and operational feedback. A metallic-fueled version of the ESFR is being evaluated in the context of the ESFR-SIMPLE project, with the aim of challenging the commercial-size oxide-fueled design in terms of safety features and economic performance. High thermal conductivity, high fuel density, ease of fabrication, inherent safety features, high achievable burnup and fuel/coolant compatibility are among attractive features of metallic fuel.

Fuel performance assessment is an essential step towards concluding on the benefits of metallic fuel technology for the ESFR design. The goal of this paper is to report on the ongoing activities tailored to modify and adapt the European codes to simulate this technology. Three different institutions and four different codes (FRED, SIM-SFR, SAS-SFR and OFFBEAT) have been involved in the development and benchmarking of metallic fuel simulation capabilities in Europe, using SAS4A/SASSYS-1 (SAS) as reference. SAS and its fuel performance model MFUEL, developed at Argonne National Laboratory (ANL), represent the state-of-the-art code for metallic fuel simulation under steady-state and transient conditions, extensively validated with EBR-II, FFTF, and PHENIX irradiation data and separate effect eutectic formation, cladding creep rupture tests and integrated TREAT M-Series and furnace tests.

The current benchmark presents the results of the six-cycles-long steady-state simulation of two fuel rods under representative irradiation conditions for inner and outer fuel core regions. The comparison of code performance focused on safety-relevant thermo-mechanical quantities; namely fuel and cladding temperatures, fission gas release, plenum pressure, fuel expansion and cladding strain. Additionally, the inner cladding wastage, due to lanthanide migration and precipitation at the boundary between the fuel and the cladding, was considered as one of the relevant phenomena for cladding integrity and has been simulated in some of the codes. The code-to-code comparison shows satisfactory agreement for most of the metrics considered in this study. Future work will focus on resolving the remaining discrepancies in cladding strain axial profile predictions and complementing these efforts with experimental validation.

Keywords: Metallic Fuel, Sodium Fast Reactors, Generation IV, Fuel Performance, Benchmark

1. INTRODUCTION

1.1. The ESFR-SIMPLE Project

The European Sodium Fast Reactor – Safety by Innovative Monitoring, Power Level flexibility and Experimental Research (ESFR-SIMPLE) Project was launched in 2022 as the latest iteration on the sodium fast reactors (SFR) design and safety assessment in Europe [1]. Building upon the experience of previous endeavors [2,3], the project strives to challenge the traditional ESFR design through the exploration of multiple technical solutions for the reactor core, the secondary system and the safety and monitoring devices. Among the current research efforts, a metallic-fueled version of the 3600 MWth ESFR core is being evaluated (ESFR-M), with the goal of offering an alternative to the commercial-size oxide-fueled design (ESFR-MOX), particularly regarding safety features and economic performance. ESFR-M features 504 fuel assemblies divided in two radial core regions (inner and outer fuel) [4], with 6 batches per region to smoothen reactivity variations and power peaking between batches. The main differences with respect to ESFR-MOX fuel and assembly design are summarized in Figure 1.

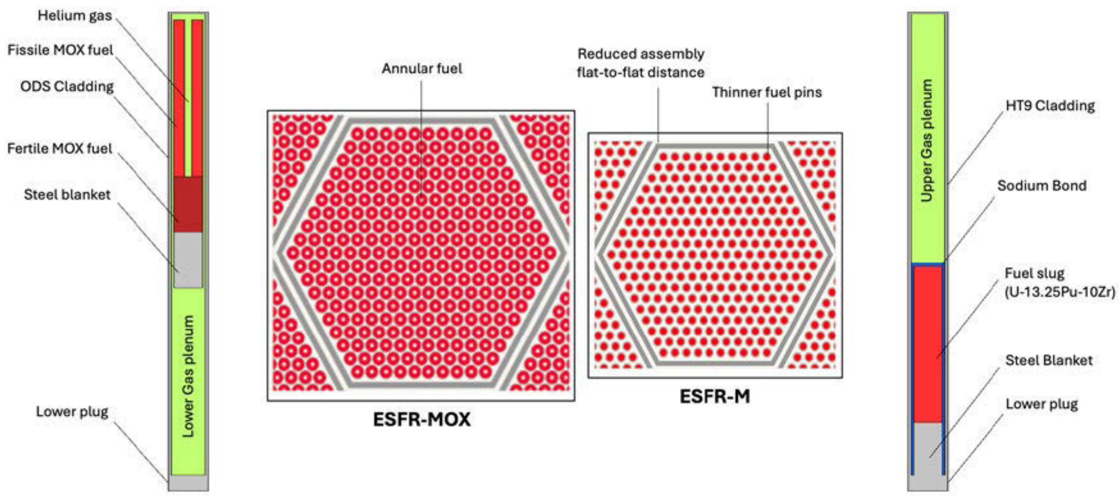

Figure 1. Main differences between ESFR-MOX and ESFR-M fuel and assembly design.

1.2. The rationale for metallic fuel in sodium fast reactors

The development of metallic fuel in Europe was discontinued in the 1960s due to concerns over irradiation-induced swelling, which was seen as a critical barrier to achieving high burnup without compromising cladding integrity [5]. However, renewed interest emerged in the 1980s through further testing conducted under the Integral Fast Reactor (IFR) Program, particularly with the Experimental Breeder Reactor II (EBR-II) [6]. These efforts revealed the previously untapped potential of metallic fuel. The case for metallic fuel in SFRs is supported by multiple advantageous features: its ease of fabrication through injection casting [7]; the high fuel density, allowing high breeding ratios with a harder neutron spectrum in a compact core; the high thermal conductivity and low specific heat, providing inherent passive safety characteristics during uncontrolled transients at high temperatures; the compatibility with the sodium coolant, resulting in no fuel loss in case of cladding breach [8]. Indeed, one of the most compelling arguments for continued investment in this technology was its performance during transient events such as Unprotected Loss of Flow (ULOF), Unprotected Loss of Heat Sink (ULOHS) and Unprotected Transient Overpower (UTOP) [9].

The introduction of this fuel solution in the ESFR-SIMPLE project required the ability to assess metallic fuel behavior under various operating conditions. Three different institutions and four different codes

(FRED, SIM-SFR, SAS-SFR and OFFBEAT) have been involved in the development and benchmarking of metallic fuel steady-state simulation capabilities in Europe, using SAS4A/SASSYS-1 (SAS) as reference. The next subsection presents the codes involved in this study.

1.3. Codes and participants

1.3.1. SAS4A/SASSYS-1

SAS4A/SASSYS-1 (SAS in the following) is a reactor dynamics and safety analysis code for liquid-metal cooled nuclear reactors under continuous use and development at Argonne National Laboratory (ANL) [10]. The fuel characterization models of SAS have recently been updated to include a new U-Pu-Zr metal fuel model, MFUEL. MFUEL features mechanistic physics-based models for base irradiation and transient simulations, accounting for all the safety-relevant phenomena occurring in a metallic-fueled core; namely zirconium redistribution, cladding wastage, fuel swelling and fission gas release, sodium relocation and fuel anisotropic axial elongation [11]. MFUEL represents the state-of-the-art model for metallic fuel simulation, extensively validated with EBR-II, FFTF, and PHENIX irradiation data and separate effect eutectic formation, cladding creep rupture tests and integrated TREAT M-Series and furnace tests [12, 13]. Thus, SAS was used as the reference code for the benchmarking activity presented in this work.

1.3.2. FRED

FRED code is under development at Paul Scherrer Institute (PSI) for the simulation of fast and LWR reactor fuel behavior under base-irradiation and accident conditions. The current version of the code solves the thermo-mechanical problem providing temperature distribution in the fuel rod and stress-strain condition of the fuel and the cladding with the traditional 1.5D approach. It includes models for changes to the gap conductance in open and closed gap regimes, fission gas release and plenum pressure. Strain components deriving from thermal expansion, sintering of the fuel porosity, swelling, creep, elastic and plastic deformations are accounted for. The code was evaluated using the data of the IFA-503.2 Halden tests with LWR fuel [14] and in the OECD MOX fuel performance benchmarks [15]. It was more recently employed in the ESFR-SMART project for an evaluation of gap conductance [16] and in the PuMMA project for a blind simulation of three high-plutonium MOX fuel irradiation experiments—CAPRIX, TRABANT1, and TRABANT2 [18]

1.3.3. OFFBEAT

OFFBEAT is an OpenFOAM®-based multi-dimensional fuel behavior code developed by the Laboratory of Reactor Safety (LRS) at École Polytechnique Fédérale de Lausanne (EPFL) and the Laboratory for Reactor Physics and Thermal-Hydraulics (LRT) at PSI [17]. It can simulate complex phenomena including 2D/3D effects and axisymmetric conditions, employing a finite volume method to discretize and solve its governing equations on unstructured meshes with arbitrary geometries. It features a thermal sub-solver, a mechanical sub-solver, gap/plena gas models and different phenomenological models for MOX and TRISO fuels. The code is parallelized for efficient computation and it has been validated for base irradiation and transient conditions for LWR and SFR fuel [18]. For this benchmark, OFFBEAT provided results for the comparison of the outer fuel pin thermo-chemical quantities and plenum properties. The contribution of this code will be extended to the inner fuel pin and the mechanical analysis in the foreseen future work.

1.3.4. SAS-SFR

The system code SAS-SFR is the result of long-term cooperation between scientists from the KIT (Germany), CEA, IRSN (France) and JAEA (Japan). The development started in the early eighties, taking the SAS4A code developed at ANL as a basis. The SAS-SFR code currently performs deterministic analysis for steady state power operation and accident conditions during the so-called initiation phase. The code solves for the 1.5D thermo-mechanical problem taking into account the degradation of the material thermal properties, the fuel swelling, the gap closure and the fission gas release. The validation database includes numerous in-pile experiments for MOX fuel elements [19]. SAS-SFR provided preliminary results for this benchmark, targeting inner fuel centerline temperature comparison. The contribution of this code will be extended in the foreseen future work.

1.3.5. SIM-SFR

SIM-SFR (being one of the SIM-“family” codes [20]) is a PC-based; multi node point kinetic (PK) model that describes the neutronic and thermal-hydraulic characteristics of both critical and sub-critical reactor cores. It has been used to perform the steady-state and transient analysis of numerous sodium cooled fast reactor designs during the following projects: CP-ESFR, ESNII+ and ESFR-SMART [21]. Actual plant data for PWRs, BWRs and the Superphenix (SPX1) reactor [22] have been used for the code validation. The code solves the thermal problem taking into account fuel and clad swelling, gap closure and the degradation of the material thermal properties. Fission gas release and plenum pressure data are also computed, while no solution of the stress-strain problem is available.

2. BENCHMARK STRATEGY AND SPECIFICATIONS

2.1. Metallic fuel physics: the phenomena modeled in this benchmark

Several physical phenomena limit ternary U-Pu-Zr metallic fuel performance during base irradiation [11]; Figure 2 displays the most relevant ones. At low burnup, early cracking is observed in high Pu-content fuel, leading to soft contact with the cladding and to restrained fuel axial expansion (Figure 2A). Solid and gaseous fission product swelling cause gap closure (Figure 2B), while the sodium bond infiltrates in the fuel open porosity, affecting its thermal conductivity (Figure 2D). As a result of a thermochemical process, Zirconium relocates inside the fuel slug, typically creating two Zr-enriched regions in the center and in the periphery of the fuel slug (Figure 2C). Insoluble lanthanide fission products migrate too and precipitate at the boundary between fuel and cladding, contributing to inner cladding wastage (Figure 2E). Combination of lanthanide attack to the cladding inner surface and iron depletion may lead to cladding embrittlement. At high burnup, if the initial fuel smear density is not low enough (i.e. if the fuel-cladding gap is not large enough), clad breach may occur due to fuel clad mechanical interaction (FCMI). Irradiation-induced embrittlement also contribute to the cladding vulnerability (Figure 2F).

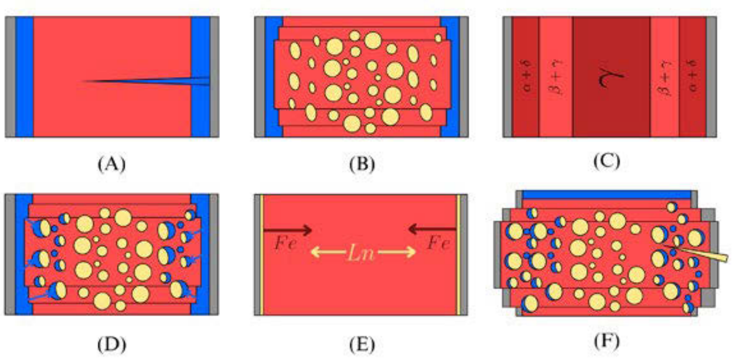

Figure 2. Physical phenomena taking place during base irradiation in metallic fuel.

The European codes involved in this task (FRED, OFFBEAT, SIM-SFR and SAS-SFR) differ significantly in terms of domains of application, numerical schemes and algorithms used for the investigation of nuclear systems. Given that the same European codes will be involved in a planned activity for system safety analysis in limiting transient conditions, this task aims at providing useful feedback on relevant quantities of interest from viewpoint of metallic fuel safety. For this reason, the adaptation of the codes to metallic fuel prioritized specific models from available literature [12]. As a first step, metallic fuel and HT9 cladding mechanical and thermal properties were implemented, accounting for irradiation-induced degradation of the fuel thermal conductivity. Furthermore, to tackle mechanical analysis and fuel-cladding mechanical interaction (FCMI), an empirical fuel axial anisotropy model and a mechanistic model to predict fission gas release and fuel swelling were adopted. Finally, to account for fuel-cladding chemical interaction (FCCI), a cladding wastage model based on precipitation kinetics [11] was included in the codes; the feedback of this model on the cladding mechanical performance being currently neglected. Zirconium redistribution and sodium infiltration models, under development in FRED and OFFBEAT codes, are not included at this stage of the comparison. Their feedback on fuel thermal conductivity is considered through a simplified burnup-based correction. The implementation of these models introduces simulation capabilities to predict thermal, mechanical and chemical quantities of interest for fuel and cladding safety.

2.2. Physical phenomena and relevant quantities for fuel and cladding integrity

In steady state conditions, the main failure criteria are related to inner cladding wastage, cladding hoop strain and plenum pressure. In order to prevent cladding failure, some recommendations were followed when designing the ESFR-M core, as summarized in Table I and detailed in previous works [4].

Table I. Link between physical phenomena, design recommendations and failure criteria.

Physical Phenomena	Failure Criteria	Design Recommendation
Zirconium redistribution and FCCI	Steady-State cladding wastage below 25% of cladding thickness	As fabricated Zr weight fraction equal to 10 % to rise solidus temperature and inhibit cladding wastage.
Fuel swelling and FCMI	Cladding hoop strain below 3%, cladding thermal creep below 1%, Cumulative Damage Fraction (CDF) below 0.05	Smeared density equal to 75% or lower to allow open porosity formation prior to hard contact between fuel and cladding.
Fission gas release	Plenum pressure below 6-7 MPa	Plenum to fuel volume ratio equal to 1.4 or higher to prevent high plenum pressure during unprotected transients.

Alongside the quantities presented in Table I, the benchmark targeted temperature distributions within the fuel slug, relevant for fuel melting margins evaluation and for Doppler feedback modeling during transients. Fuel axial expansion is also included in the comparison, given its relevance for core reactivity both in steady-state and transient conditions.

2.3. Benchmark specifications

Table II details the specifications for the two fuel pins considered in this work. Inner (IF) and outer fuel (OF) average conditions were simulated, considering irradiation histories and axial profiles as obtained from a six-batch-long full core neutronic assessment of the ESFR-M, using the Monte Carlo code for neutron transport Serpent [4]. The relevant power information is shown in Figure 3; it reflects the saw-shaped reactivity swings obtained with a six-batch reloading scheme and the axial power profile, assumed time-independent for these simulations. The main differences between the two considered pins consist in their active height and in their linear power. The fast-flux-to-linear-power-ratio is similar for the two fuel cases and it is used in SAS for first-order approximation of the cladding dose (in dpa) attained during irradiation. The SAS simulations used as reference for this study are discussed in detail in [23].

Table II. Benchmark Specifications.

	Inner Fuel (IF)	Outer Fuel (OF)
Fuel Material	U-13.25Pu-10Zr	U-13.25Pu-10Zr
Fuel Reference Density (g/cm^3)	15.80	15.80
Fuel Slug Radius (cm)	0.3863	0.3863
Cladding Material	HT9	HT9
Cladding Reference Density (g/cm^3)	7.63	7.63
Cladding inner radius (cm)	0.4501	0.4501
Cladding outer radius (cm)	0.5025	0.5025
Mass of Na in the gap (g)	13.02	16.18
Active fuel height (cm)	75	95
Gas plenum height (cm)	120	100
Coolant inlet temperature (K)	633	633
Flowrate per pin (kg/s)	0.162	0.120
Hydraulic diameter (cm)	3.89×10^{-1}	3.89×10^{-1}
Flow area per pin (cm^2)	3.30×10^{-1}	3.30×10^{-1}
Nominal Assembly Power (MW)	8.354	6.235
Linear Power (W/cm)	411.02	242.18
Fast-flux-to-linear-power ratio (1/m-W-s)	6.13×10^{14}	6.22×10^{14}
Dose conversion factor (dpa·cm^2·s)	4	4

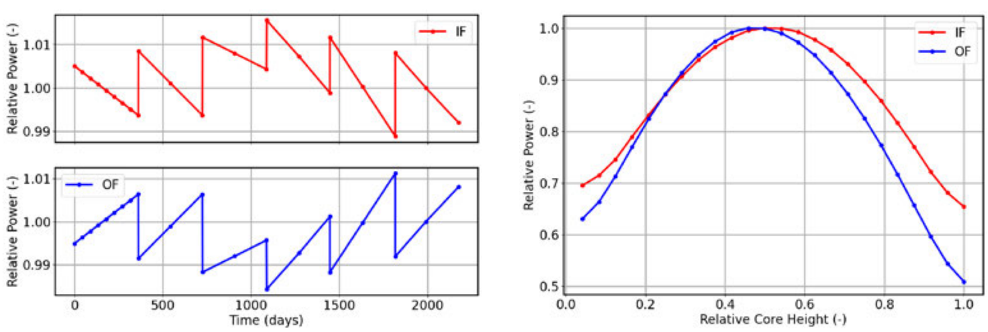

Figure 3. Power evolution (on the left) and axial profile (on the right) for IF and OF fuel.

3. BENCHMARK RESULTS

Most of the codes involved in the task are in good agreement with the reference SAS simulations for inner and outer fuel average conditions. Fuel centerline temperature profiles are within 30°C difference during irradiation as shown in Figure 4 for FRED, SIM-SFR and OFFBEAT. The comparison deteriorates at higher burnups as a result of the simplified thermal conductivity model adopted by the European codes (see SAS-SFR as an example). Future efforts shall focus on improving these predictions using the zirconium redistribution and sodium infiltration models. As shown in Figure 4, occasionally fuel temperature predictions of SAS can have slight discontinuities. This is due to phase dependent behavior of zirconium redistribution, sodium infiltration and a coarse radial nodalization scheme. Discontinuities is more of an issue for high Pu containing fuels with formation of complex phase structures and fuel constituent redistribution.

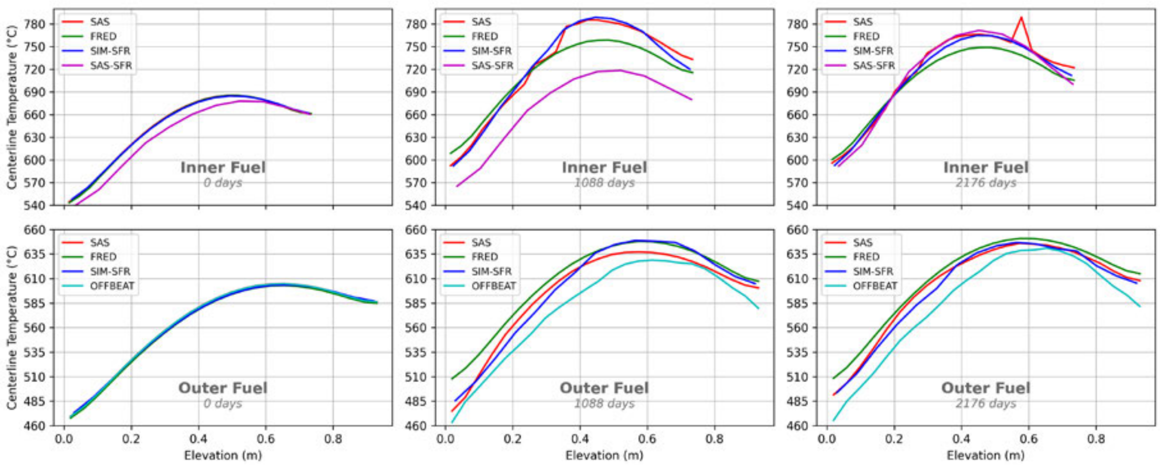

Figure 4. Thermal analysis: fuel centerline temperature axial profile along the irradiation.

Peak cladding wastage is well predicted by the precipitation kinetics model adopted by the codes shown in Figure 5. The comparison is satisfactory for both inner and outer fuel. As discussed in [23], it remains well-below the safety limit of 25% of the as-fabricated cladding thickness.

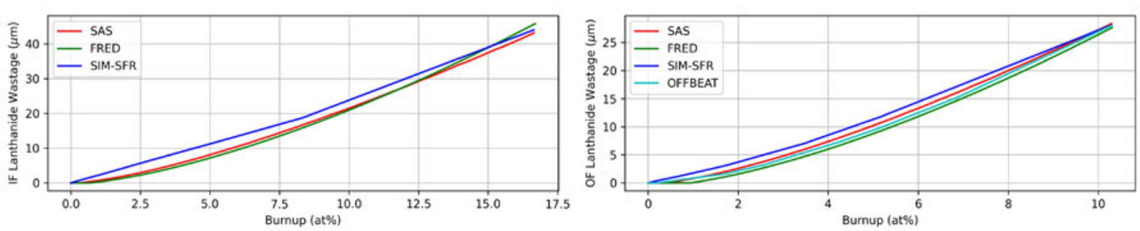

Figure 5. Chemical analysis: lanthanide cladding wastage for IF (left) and OF (right).

The approaches to the solution of the mechanical problem are different in FRED and SAS. SAS/MFUEL employs a reduced-order-model to compromise between numerical stability and accuracy, while FRED solves for the stress-strain profile inside pin and cladding. The latter approach results in some instabilities, due to the high creep rate of metallic fuel, which currently limit the performance of FRED. Current efforts are devoted to increasing the numerical stability of the code. For the considered fuel pin, fuel axial expansion and cladding hoop strain axial profile at the end-of-life (EOL) exhibit a satisfactory agreement as shown in Figure 6. Cladding hoop strain remains below the 3% safety limit, peaking at the bottom of the slug due to the lower sintering rates of the fuel open porosity.

The comparison of relevant plenum properties such as fission gas release (FGR) and plenum pressure is shown in Figure 7. Although FRED predicts a delayed onset of FGR relative to the reference SAS results, it captures the overall trend and the final release value at the end of irradiation for both inner and outer fuel conditions. In addition to the previously discussed differences in the mechanical solvers, the discrepancies observed may also stem from variations in the implementation of the FGR model. For instance, the mechanistic model in FRED categorizes fission gas bubbles into two groups based on size, whereas SAS utilizes a three-group classification scheme. The plenum pressure increase, caused by the simultaneous effect of FGR and sodium bond relocation above the fuel slug, is perfectly captured by the codes involved in this task. The accurate prediction of this quantity during base irradiation provides relevant feedback for the analysis of limiting transients. In ULOF and UTOP, at elevated temperatures, the fuel becomes softer and forms a eutectic with the cladding [11]. Thus, FCMI becomes negligible and plenum pressure becomes critical for cladding safety, as it drives cladding hoop stress.

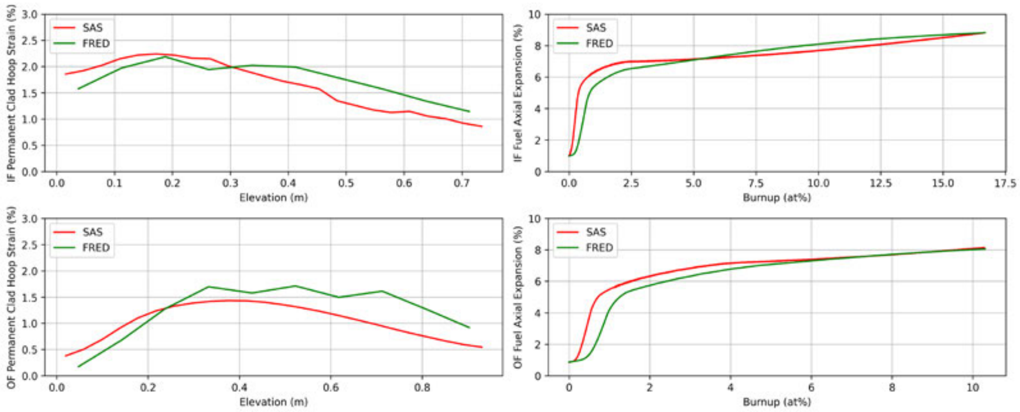

Figure 6. Mechanical analysis: permanent cladding hoop strain and fuel axial expansion for IF (above) and OF (below).

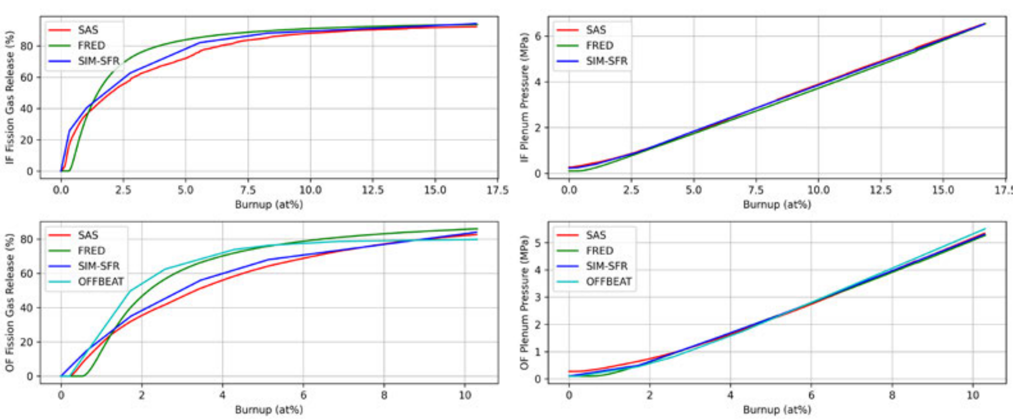

Figure 7. Plenum properties: fission gas release and plenum pressure for IF (above) and OF (below).

4. CONCLUSIONS

This benchmark represents an important initial step toward establishing metallic fuel simulation capabilities in Europe. It targeted the six-cycles-long steady-state simulation of two fuel rods under representative irradiation conditions for inner and outer fuel core regions. The code-to-code comparison involved three institutions and four codes, benefiting from the availability of reference simulations using the state-of-the-art code SAS. The benchmark demonstrated satisfactory agreement for most of the metrics considered in this study, including, thermal, chemical and mechanical quantities. Future efforts will focus on completing the benchmark comparison for some of the codes involved in this task (SAS-SFR and OFFBEAT), and addressing the remaining discrepancies in the mechanical problem solutions. These developments will be supported by validation against experimental data.

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