Nuclear Engineering and Design 420 (2024) 113048

Contents lists available at ScienceDirect

Nuclear Engineering and Design

journal homepage: www.elsevier.com/locate/nucengdes

Performance analysis and helium behaviour of Am-bearing fuel pins for irradiation in the MYRRHA reactor

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A B S T R A C T

Minor actinides are the main contributors to medium- and long-term radiotoxicity and heat production in spent nuclear fuels. Research efforts are currently ongoing to explore different options to dispose of such radionuclides, e.g., their burning in fast reactors within mixed-oxide fuels. The MYRRHA sub-critical reactor is one of the future facilities with envisaged burning and transmutation capabilities. This work assesses the thermal–mechanical performance of a homogeneous Am-bearing fuel pin both in the In-Pile test Section position of the MYRRHA "Revision 1.8" core and under driver irradiation. The normal operating conditions of MYRRHA are considered, with a focus on the safety design limits and involving sensitivity analyses to evaluate the impact of increasing americium contents (in the range 0–5 wt%) on safety-relevant simulation outcomes. The simulations are performed with the TRANSURANUS fuel performance code (version v1m4j22) coupled with the SCIANTIX physics-based module for inert gas behaviour, and rely on a dedicated surrogate model for the helium source term during MYRRHA irradiation accounting for the relevant contribution of the fuel americium enrichment, besides advanced models for the properties and behaviour of the specific pin materials. The analyses reveal the suitability and safety under irradiation of MOX fuels with low Am enrichments according to the current MYRRHA design.

1. Introduction

Our current society is in the process of transitioning to a more sustainable global system, and while the task ahead is extremely challenging and involves all the human activities, a significant fraction of this effort is related to energy (Schlindwein and Montalvo, 2023). Nuclear energy can provide a significant contribution to the global effort of phasing out of fossil fuels, but it is still not widely accepted by the population. One of the most recurring themes that hinders nuclear social acceptance is the production and management of long-lived radioactive nuclear waste (Salvatores, 2000), which is why research on innovative, advanced, and safe methods to dispose this hazardous material is still ongoing (Mukaiyama, 1994; Weiss et al., 2013; Merk et al., 2017). Indeed, the current fleet of light water reactors produces considerable amounts of high-level waste, and while most of it (U and Pu) can be partially recycled as fuel via e.g., advanced fuel cycle concepts as the twice-through one (Poinssot et al., 2012); (Rodriguez-Penalonga and Moratilla Soria, 2017), a fraction of the remaining minor actinides is still a significant contributor to long-term radiotoxicity. The feasibility of the Partitioning and Transmutation (P&T) route for these radionuclides through the use of fast reactors (FR) is being explored (Salvatores et al., 1998; Mueller, 2013; Fujimura et al., 2015). Advanced FR systems are a sub-set of the Generation IV reactor concepts (GIF (Generation IV International Forum), 2018); (GIF (Generation IV International Forum), 2021), which comprehend liquid metal-cooled systems featured by an improved sustainability, i.e., a more efficient fuel utilization and minimization of the residual waste from nuclear applications (Locatelli et al., 2013); (GIF (Generation IV International Forum), 2019).

The PATRICIA (Partitioning And Transmuter Research Initiative in a Collaborative Innovation Action) H2020 Project (European Union's Horizon, 2020) focuses on developing advanced partitioning techniques to efficiently separate americium from spent fuel and to analyse the behaviour under irradiation of Am-bearing nuclear fuel. In this framework, this work deals with the performance assessment of an Am-MOX pin, i.e., fuelled with (U, Pu, Am)O_{2-x} and designed for transmutation purposes, for future irradiation in the MYRRHA reactor concept (Abderrahim et al., 2019). MYRRHA is an experimental FR concept for research applications, cooled by Lead-Bismuth Eutectic (LBE) and equipped with an external accelerator (Accelerator Driven System, ADS) coupled to the sub-critical core configuration. Indeed, the MYRRHA reactor, being designed by SCK CEN (Belgium), will have two operating modes: the sub-critical mode and the critical mode. The first one will require a continuous injection of high-energy neutrons, which will be provided via a LINAC that is being built at the reactor site. This machine

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https://doi.org/10.1016/j.nucengdes.2024.113048
Received 28 July 2023; Received in revised form 13 December 2023; Accepted 21 February 2024
Available online 29 February 2024

will be capable of accelerating protons up to 600 MeV; these high energies enable spallation reactions on heavy atoms such as the lead and bismuth of the coolant, producing multiple instances of high-energy neutrons. The target of this particle beam will be a channel of liquid LBE located in the centre of the reactor core. This device, in synergy with the LBE coolant, will produce the hardest neutron spectrum ever achieved up to now in a fission reactor. High-energy neutrons are capable to directly fission fissionable nuclides, that is why one of the main goals of the MYRRHA facility is to study the P&T of minor actinides, most of which are fissionable nuclides.

The current design specifications of the sub-critical configuration of the MYRRHA reactor ("Revision 1.8", provided within the PATRICIA Project (European Union's Horizon, 2020)) foresee a thermal power output of 70 MW_{th} produced among 78 fuel assemblies, which will be loaded with a U-Pu MOX fuel encased in a DIN 1.4970 cladding, which is a specific alloy, annealed and cold-worked, of the 15-15Ti stainless steel family. This core configuration is not just composed of fuel assemblies and the central LBE spallation target, but it comprehends some experimental assemblies in the second core ring dedicated to irradiation experiments (IPS - In-Pile test Sections), in which performance testing of Am-MOX pins is envisaged, besides thermal islands for thermal neutron irradiations (ThIPS) and safety rods (CR) in the fourth core ring (Fig. 1). Moreover, the core is surrounded by a stainless steel jacket and includes MgO reflector channels and LBE channels at the periphery. The analysis of the IPS irradiation is particularly of interest to understand the behaviour of Am-MOX fuel during a single MYRRHA cycle, to support the qualification of these pins as driver (or blanket) fuel towards advanced configurations of the reactor core devoted to transmutation / burning goals.

The analyses presented in this work are performed using the TRANSURANUS fuel performance code (FPC) (Magni et al., 2021; Lassmann, 1992), coupled with the SCIANTIX module for inert gas behaviour and release modelling (Pizzocri et al., 2020; Zullo et al., 2023; Van Uffelen et al., 2020; Pizzocri et al., 2021) in support of the design and performance studies of MYRRHA Am-MOX pins. The focus of this work is on the pin safety under irradiation, determined by comparison with design limits, as well as on the capabilities of burning americium and on the production of helium in Am-MOX fuels. For this purpose,

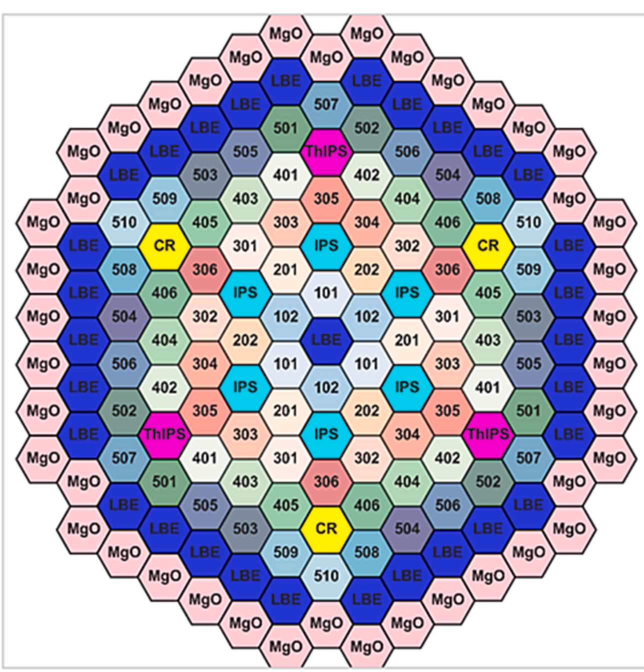

Fig. 1. Scheme of the current MYRRHA "Revision 1.8" core design, sub-critical configuration.

SCIANTIX has been equipped with a surrogate model for He production dedicated to the MYRRHA fuel composition and irradiation conditions. This model is built on synthetic datasets of produced He generated via the burnup module capabilities of SCIANTIX itself (Cechet et al., 2021; Pizzocri et al., 2023), fed by neutronic data (cross-sections, besides fission yields and decay constants) derived from a SERPENT modelling and simulation of the MYRRHA pin.

The paper is organized as follows. Section 2 details the specifications of the MYRRHA irradiation scenarios considered in this work, while Section 3 presents the simulation tools and modelling set-up (including the MYRRHA-oriented He production model) applied to analyse the Am-MOX pin performance. The code results are provided in Section 4 for both the IPS and driver fuel irradiation scenarios, complemented by sensitivity analyses focusing in particular on the impact of the fuel composition in terms of Am content. Conclusions and further developments following the present work are drawn in Section 5.

2. Specifications of the MYRRHA irradiation scenarios

The focus of the analyses performed in this work is the irradiation (in the current MYRRHA "Revision 1.8" core design) of a MOX fuel pin loaded with americium, either for 90 days of single-cycle, full power operation within the IPS experimental assembly or as a driver pin experiencing the complete in-reactor history from first loading to discharge at the target burnup. The IPS experiment is indeed mainly designed to qualify Am-MOX pins as MYRRHA driver fuel, i.e., fuel pins irradiated for an extended period of 13 irradiation cycles, each one corresponding to an irradiation position along the MYRRHA core (from the hottest one, in the first core ring closer to the central spallation channel, to the last one in the peripheral fifth core ring). The handling of the fuel assemblies (FAs) follows the reshuffling scheme illustrated in Fig. 2a (where the numbers 1 and 13 correspond to the initial and final positions of the driver FA path during its life in the core) [1] (Magni et al., 2022). Each of the irradiation cycles lasts 90 days as the IPS irradiation. The IPS irradiation position is placed in the second ring of the MYRRHA core, just like the second irradiation position for the driver fuel. More specifically, the neutron flux (and consequent linear power produced) that a fuel pin experiences in the IPS is assumed as the average between the two surrounding, second-ring driver positions, i.e., number 201 and 202. For what concerns the thermal boundary conditions in the IPS, its primary coolant-cooled configuration is considered in this work (i.e., same LBE inlet temperature and mass flow rate designed for the driver assemblies), while the possibility of an isolated IPS configuration is still under consideration for MYRRHA.

The irradiation conditions shown in Fig. 2b correspond to the "hottest" Am-MOX pin, i.e., to the conservative assumption of considering the pin placed in the harshest position (in terms of neutron flux and linear heat rate, and consequently fuel temperature) within the fuel assembly, in each of the irradiation cycles. Hence, the performance of the hottest Am-MOX pin is analysed in this work, according to the same approach already adopted in (Magni et al., 2022; Magni et al., 2023) and applied to MYRRHA U-Pu MOX fuel. This is important considering that a scope of this analysis is to provide indications about the safety of Am-loaded pins under the normal operating conditions designed for the MYRRHA "Revision 1.8" reactor.

For what concerns the fuel composition, the primary option for the current MYRRHA core design foresees an amount of plutonium and americium which correspond together to 30 wt.% of the heavy metals in the Am-MOX fuel. In this study, the homogeneous strategy for the

[1] Considering the MYRRHA core symmetry, the driver fuel irradiation is performed in each one-sixth of the core simultaneously. With reference to the core portion highlighted by the red triangle of Fig. 2a, the driver fuel irradiation progressively occupies the positions 102-202 – 302-304 – 306-402 – 404-406 – 502-504 – 506-508 – 510.

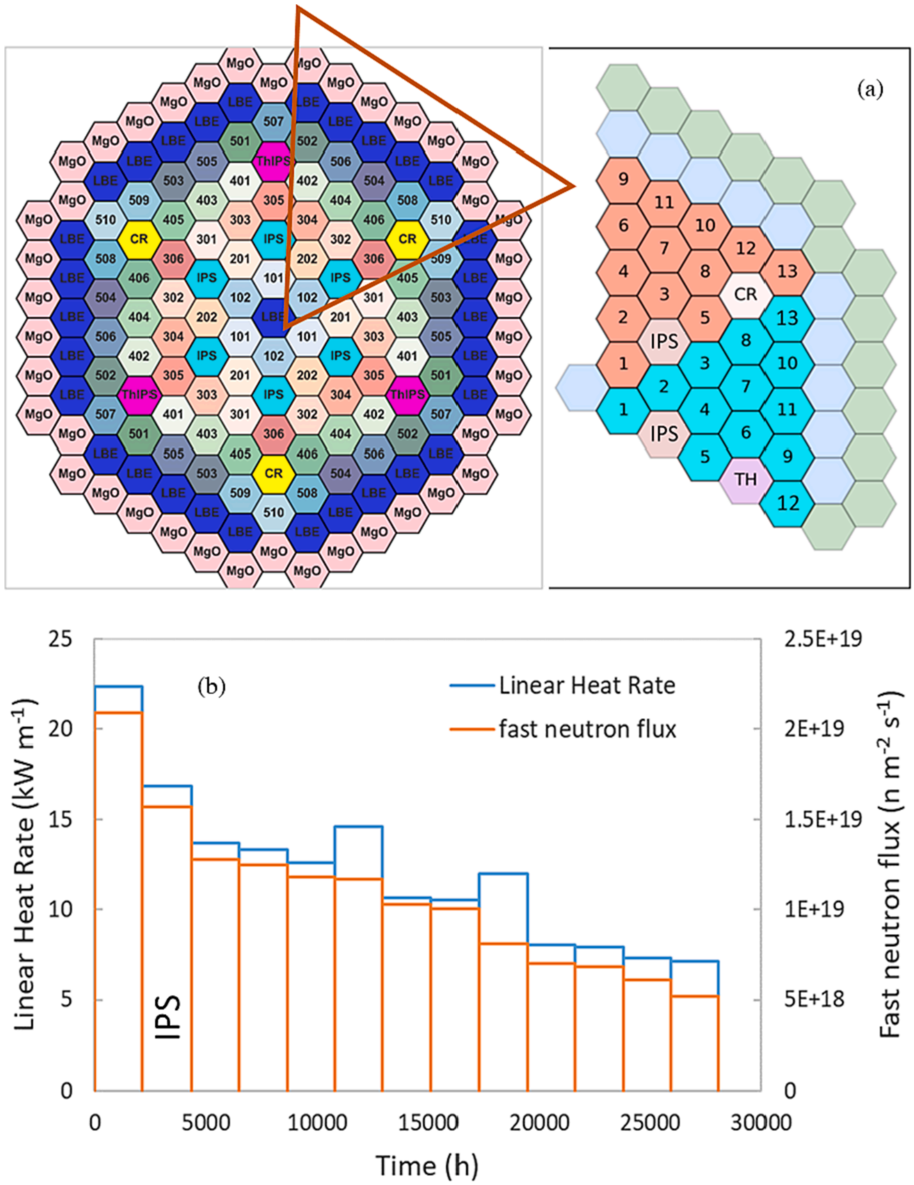

Fig. 2. (a) Scheme of the 13 irradiation positions during the normal operation of the MYRRHA driver fuel (Magni et al., 2022). (b) Irradiation conditions of the hottest pin under both IPS and driver scenarios, at the peak power node, corresponding to the "Revision 1.8" specifications of the MYRRHA reactor core. The values shown correspond to the reference Am-MOX composition designed for MYRRHA, i.e., 0.49 wt% Am and 29.51 wt% Pu.

recycling and transmutation of americium is analysed: the refence Am content is set to 0.49 wt.% and the impact of a 5% Am content (limit for the homogeneous strategy) on the pin performance is showcased. The irradiation history (both linear power and fast neutron flux) shown in Fig. 2b holds for the reference 0.49 wt.% Am content. The actual neutron fluxes and powers in both the IPS single-cycle and each of the 13 driver irradiation cycles would depend on a different fuel Am content, besides on e.g., the experimental rig and irradiation environment in the IPS position. The assumption adopted in this work is that Am contents in the range 0.49 – 5 wt.% have a negligible impact on the MYRRHA irradiation conditions, otherwise additional neutronic computations specific for each fuel composition (out of the scope of this work) should support the pin performance analyses.

For what concerns the main specifications related to core and pin geometries, and as-fabricated material parameters, they are mostly in line with the previous MYRRHA design "Revision 1.6", already targeted by (Magni et al., 2022; Magni et al., 2023). They are recalled and collected in Table 1 since MYRRHA "Revision 1.8" is the latest, updated core design targeted by this work and of reference within the PATRICIA Project. The main differences with the previous "Revision 1.6" design consist in the presence of the IPS irradiation position within the second core ring, the Am-bearing mixed-oxide fuel instead of U-Pu MOX, the lower linear heat rates and the lowered coolant inlet temperature (from 245°C to 220°C).

3. Modelling and simulation set-up

The simulation tool employed for the present pin performance analyses is the coupled suite between the TRANSURANUS 1.5D fuel performance code (version v1m4j22) (Magni et al., 2021; Lassmann, 1992), working at the integral scale of the fuel pin for thermal–mechanical evaluations, and the SCIANTIX grain-scale code (originally developed by Pizzocri et al. (Pizzocri et al., 2020)) devoted to physics-based calculations of inert gas behaviour within the fuel matrix. An advanced version (2.0) of SCIANTIX, object-oriented and with extended modelling capabilities, has been recently developed and assessed (Zullo et al.,

Table 1
Geometry specifications of the MYRRHA "Revision 1.8" core and as-fabricated Am-MOX MYRRHA fuel pins.

Parameter	Value
Assembly-level	
Number of fuel pins per assembly	127
Pin pitch (hexagonal lattice)	8.4 mm
Total coolant mass flow rate	71.4 kg s^{-1}
Coolant mass flow rate (pin-centred) [a]	0.48 kg s^{-1}
Coolant passage area (equivalent, pin-centred) [a]	25 mm^2
Coolant inlet temperature	220°C
Coolant inlet pressure	0.6 MPa
Pin-level	
Fuel column length	650 mm
Wire-spacer diameter	1.8 mm
Upper plenum length	60 mm
Lower plenum length	580 mm
Fill gas temperature	20°C
Fill gas pressure	0.1 MPa
Fuel outer diameter	5.42 mm
Cladding inner diameter	5.65 mm
Cladding outer diameter	6.54 mm
Fuel specifications	
Am (reference) concentration	0.49 wt.%
Pu (reference) concentration	29.5 wt.%
U-235 enrichment / U (natural U)	0.711 wt.%
Fuel grain size	10 µm
Fuel oxygen-to-metal ratio	1.969
Fuel density	95% TD
Fuel operation	
Average linear power (IPS)	15.24 kW m^{-1}
Maximum linear power (IPS) [a]	16.87 kW m^{-1}
Average linear power (first cycle)	19.45 kW m^{-1}
Maximum linear power (first cycle) [a]	22.35 kW m^{-1}

[a] The values hold for the hottest pin, supposed to cover "interior" positions within the assemblies (avoiding corner or edge effects), in both the IPS and during each of the 13 cycles of driver fuel operation.

2023) and is herein applied.

The modelling set-up employed for the performance analyses of Am-MOX fuels under MYRRHA irradiation scenarios (Section 2) is the same as in a recent work targeting the behaviour of U-Pu MOX fuel under MYRRHA transient conditions (Magni et al., 2023). In addition to this, a model dedicated to the helium production in MYRRHA Am-MOX, relevant due to the fuel Am content herein considered, is adopted. These models are recalled / developed in the following sub-sections.

3.1. Thermal-mechanical properties and behavioural models

The TRANSURANUS code has been equipped with advanced models of thermal–mechanical properties, suitable for the MYRRHA pin materials (fuel, cladding, coolant) and irradiation conditions, which are here recalled and referenced. First, the models for the thermal properties (thermal conductivity, melting temperature) of Am-bearing oxide fuels proposed and validated in (Magni et al., 2021) are used, covering the ranges associated to the MYRRHA irradiation in terms of temperature, Am and Pu contents, deviation from stoichiometry, porosity and burnup (relevant especially for the driver fuel irradiation), and providing explicit dependencies on the fuel Am content. Second, recent recommendations for fuel mechanical properties (thermal expansion, elastic moduli), suggested by (Lemehov, 2020), are employed for the simulations. These models are actually developed for U-Pu MOX, and are still selected in light of the low (homogeneous, within [0 ÷ 5] wt.%) Am concentration in the as-fabricated fuel currently considered for MYRRHA applications. This results in a limited impact on the mechanical properties compared to U-Pu MOX (Sobolev et al., 2003; Kato et al., 2011; Prieur et al., 2015), hence the models for MOX elastic moduli and strain due to thermal expansion can be deemed suitable also for Am-MOX (while a dedicated modelling should be necessary to target oxide fuels bearing higher amounts of minor actinides, for e.g., blanket fuel and transmutation purposes (D'Agata et al., 2017)). This is in line with the state-of-the-art approach for the modelling available in fuel performance codes (Van Uffelen and Suzuki, 2012; Van Uffelen et al., 2019), and the same strategy is adopted for all the other, several fuel properties not mentioned before (e.g., specific heat, creep).

For what concerns the cladding modelling, the standard TRANSURANUS models for the properties of 15-15Ti steels are adopted, apart from models developed specifically for the MYRRHA cladding steel (DIN 1.4970, of the 15-15Ti steels family) and published in (Magni et al., 2022). These models concern the thermal and irradiation-induced creep, void swelling and time-to-rupture, applicable to ranges relevant for the current MYRRHA core design. Advanced models are also implemented in TRANSURANUS and employed for the thermophysical properties of the LBE coolant (e.g., thermal conductivity, specific heat, viscosity), aligned with the recommendations provided by the latest NEA Handbook (OECD/NEA, 2015). The heat transfer coefficient between cladding and coolant is modelled via the Ushakov correlation (Ushakov et al., 1977), of reference in TRANSURANUS for lead / bismuth / LBE coolants. Additional correlations suitable for LBE (i.e., those by Subbotin and Kazimi-Carelli, reviewed by (Mikityuk, 2009) and recently by (Di Gennaro et al., 2023)) prove to be more conservative in the MYRRHA normal operation conditions since providing a lower Nusselt number (~20–25% lower). This range corresponds to the variation of the LBE Nusselt number considered in the sensitivity analysis performed in Section 4.3.

The most important advancement concerning behavioural models consists in the use of SCIANTIX (version 2.0 (Zullo et al., 2023)) as a physics-based module dedicated to inert gas (xenon, krypton, and helium) behaviour coupled to TRANSURANUS. This enables a coherent calculation of fuel swelling and gas release [2] in the fuel-cladding gap resulting from the intra- and inter-granular description of the gas dynamics accounting for lower-length scale data and information embedded in the model parameters (e.g., diffusivities, trapping and re-solution rates, fractional coverage of the grain boundaries) (Zullo et al., 2023; Pizzocri et al., 2022; Pizzocri et al., 2022). In this way, the FPC benefits from the bridging with the atomistic scale according to a consistent multi-scale framework, and overcomes the correlation-based modelling typically adopted by engineering softwares. The physics-based nature of the SCIANTIX modelling, with parameters adjustable to account for specificities of different types of oxide fuels, guarantees the general validity and applicability of the models also to the Am-bearing oxide fuels designed for MYRRHA.

When running the TRANSURANUS//SCIANTIX coupled code suite, SCIANTIX is called by TRANSURANUS at every radial / axial node of the adopted fuel discretization mesh, within each calculation loop that the FPC performs at each time step. This happens online during the simulation, according to a coupling methodology and interface developed in (Van Uffelen et al., 2020; Pizzocri et al., 2019). The coupling between TRANSURANUS and SCIANTIX is already demonstrated and assessed in previous works from the same authors, concerning the application of the coupled suite to both water-cooled and liquid metal-cooled irradiation experiments and reactor designs (Pizzocri et al., 2021; Magni et al., 2023; Zullo et al., 2022; Zullo et al., 2023; Luzzi et al., 2023; Magni et al., 2022).

[2] The release of both helium and fission gases from the fuel grain boundaries to the pin free volume, during the coupled TRANSURANUS//SCIANTIX simulation, is handled by SCIANTIX based on the evaluation of the fractional coverage of the grain boundaries by inter-granular bubbles. When this fractional coverage overcomes a grain boundary saturation threshold, the excess gas is released so that the equilibrium condition is restored (Zullo et al., 2023; Pastore et al., 2013).

3.2. Surrogate model for helium production

The SCIANTIX code, herein applied as an inert gas behaviour module coupled to TRANSURANUS, provides stand-alone burnup module capabilities originally developed and verified for FR-type U-Pu MOX by (Cechet et al., 2021), then extended and applied to Am-MOX under FR conditions in (Pizzocri et al., 2023). The burnup module relies on an offline-online methodology, consisting of a first offline step of derivation of nuclear data (reaction cross-sections) specific for a certain reactor-fuel couple via SERPENT (Leppänen, 2013; Leppänen et al., 2015) simulations spanning the ranges of power, neutron flux, temperature and fuel composition of interest [3]. The resulting matrices of average cross-sections [4] are implemented in the SCIANTIX burnup module routines, to allow the online solution of the Bateman equations related to the nuclides of interest, i.e., the isotopes of uranium, plutonium and minor actinides, and helium (accounting for the three main ways of He production, i.e., α-decays of actinide isotopes, ternary fissions and (n,α) reactions).

The same approach followed in previous works by the same authors (Cechet et al., 2021; Pizzocri et al., 2023) is adopted here to derive an accurate surrogate model of He production in Am-MOX fuels under MYRRHA irradiation conditions, based on multiple SERPENT-informed SCIANTIX runs generating synthetic datasets of values of He produced during the involved irradiation conditions (power, neutron flux and irradiation time leading to a certain fuel burnup) and for different fuel initial enrichments. The modelling activity focuses here on the IPS irradiation position, but the verified ranges of applicability of the developed He production model are wide enough to cover the whole irradiation conditions of MYRRHA driver fuel, both axially along the pin active length and across the core, from the first loading in the first core ring (the closest to the central spallation target) up to the last irradiation cycle at the core periphery before fuel discharge (Section 2). Moreover, the aim is to develop a model providing the explicit dependency on the fuel Am content, hence enabling to account for this relevant effect on the He production in a reliable but engineering way, and to isolate it for dedicated uncertainty and sensitivity analyses.

The results of the SCIANTIX burnup module on the MYRRHA Am-MOX (at the reference Am content of 0.49 wt.% and at 5 wt.% Am), working with the cross-sections customized for the MYRRHA reactor, are verified against the high-fidelity results of the SERPENT simulations and of TUBRNP (the TRANSURANUS burnup module (Magni et al., 2021; Botazzoli et al., 2011)) under the same conditions, i.e., a continuous irradiation at the IPS neutron flux and power levels up to an extended burnup of 200 GWd/t_{HM}. Fig. 3 showcases the evolution with burnup of three nuclides which are particularly significant for the helium production in Am-MOX fuels, besides the helium production itself. It is worth noticing the remarkable agreement between the predictions provided by SCIANTIX and the high-fidelity values from SERPENT, resulting in an advanced and more reliable estimation of the He produced especially at higher Am enrichments. It should be recalled that TUBRNP is currently verified up to 100 GWd/t_{HM}, so its predictions shown in Fig. 3 are extrapolated out of range towards very high burnup levels (Botazzoli et al., 2011).

Based on the results of the SCIANTIX burnup module for He production, a model is developed to describe the helium production rate for the Am-MOX fuel in the MYRRHA irradiation conditions (IPS / driver). The main aim is to achieve a description capable of accounting for the actual irradiation conditions and fuel composition (especially in terms of Am) during the MYRRHA irradiation. Additionally, this surrogate model allows faster computing times, suitable for implementation in / coupling with fuel performance codes, compared to the running times of burnup modules (Cechet et al., 2021). The model derivation is performed by best-fitting the synthetic dataset of He production values provided by SCIANTIX, in wide ranges of MYRRHA-relevant irradiation conditions and fuel composition (Table 2). Statistical analyses on the p-values associated to each model regressor are performed with the R statistical code (R Foundation, 2018), to assess the significance of the proposed model formulation against the fitted data. The resulting surrogate model for the He production rate, which proved to best represent the synthetic "training" dataset derived via multiple SCIANTIX simulations, reads:

$$\begin{aligned}\frac{d[^4\mathrm{He}]}{dt} = {} & \left(A\left[^{241}\mathrm{Am}\right] + B\left[^{241}\mathrm{Am}\right]^2 + C\right) + 2\,t_{irr}\left(D\left[^{241}\mathrm{Am}\right] + E\left[^{241}\mathrm{Am}\right]^2 + F\right) + \dot{F}\left(G\left[^{241}\mathrm{Am}\right] + H\right) + 2\,t_{irr}\left(\dot{F}\right)^2\left(I\left[^{241}\mathrm{Am}\right] + J\right) \\ & - K\left[^{241}\mathrm{Am}\right]\frac{\dot{F}}{X}\left(\frac{t_{irr}\dot{F}}{X} - 1\right)e^{-\left(\frac{t_{irr}\dot{F}}{X}\right)}\end{aligned} \tag{1}$$

where the coefficient values are listed in Table 3 and hold for Am concentration [^{241}Am] expressed in wt.%, irradiation time t_{irr} in hours, and fission rate density $\dot{F}$ in fissions m^{-3} s^{-1}. This model can replicate with great accuracy the results of helium production from the SCIANTIX burnup module within the validity intervals (Table 2), i.e., the model verification against the training dataset is confirmed, as shown in Fig. 4 - left. It is also successfully validated against additional, independent datasets for which the input parameters have been sampled randomly within the same ranges of values (Table 2), as shown in Fig. 4 – right. The proposed model shows robustness also outside of its validity intervals, e.g., against datasets corresponding to up to 10 wt.% of Am, lower fission rate densities down to $0.9 \cdot 10^{19}$ fissions m^{-3} s^{-1} or up to $3.0 \cdot 10^{19}$ fissions m^{-3} s^{-1}, hence covering the whole range of irradiation conditions designed for the driver fuel (i.e., the entire pin irradiation history from cycle 1 to cycle 13) in MYRRHA "Revision 1.8". The RMSEs associated to these additional datasets are: 0.07 for the "validation" dataset of Fig. 4 – right, 0.49 for the dataset up to 10 wt.% of Am content in the fuel, 0.22 and 0.12 for the "lower power" and "higher power" datasets, respectively.

4. Fuel pin performance results

The results of the MYRRHA Am-MOX pin performance are provided and discussed in this Section, obtained by calculations performed as

[3] The geometry and input specifications employed for the SERPENT calculations performed correspond to an infinite hexagonal lattice of fuel pins with the same dimensions (e.g., pin radius, pin pitch, active length) as in the MYRRHA current design (Sections 1 and 2), as well as the same fuel, cladding and coolant compositions, while no structural materials are included in the SERPENT model.

[4] The SCIANTIX burnup module differs from TUBRNP (the burnup module in use in TRANSURANUS (Magni et al., 2021; Botazzoli et al., 2011)) since e.g., a higher number of actinide isotopes is accounted for (hence more α-emitting paths providing an advanced description of the He production evolution with burnup), and the reaction cross-sections are not constant values but each one is implemented as a matrix that depends on the fuel initial enrichment and actual burnup (Cechet et al., 2021).

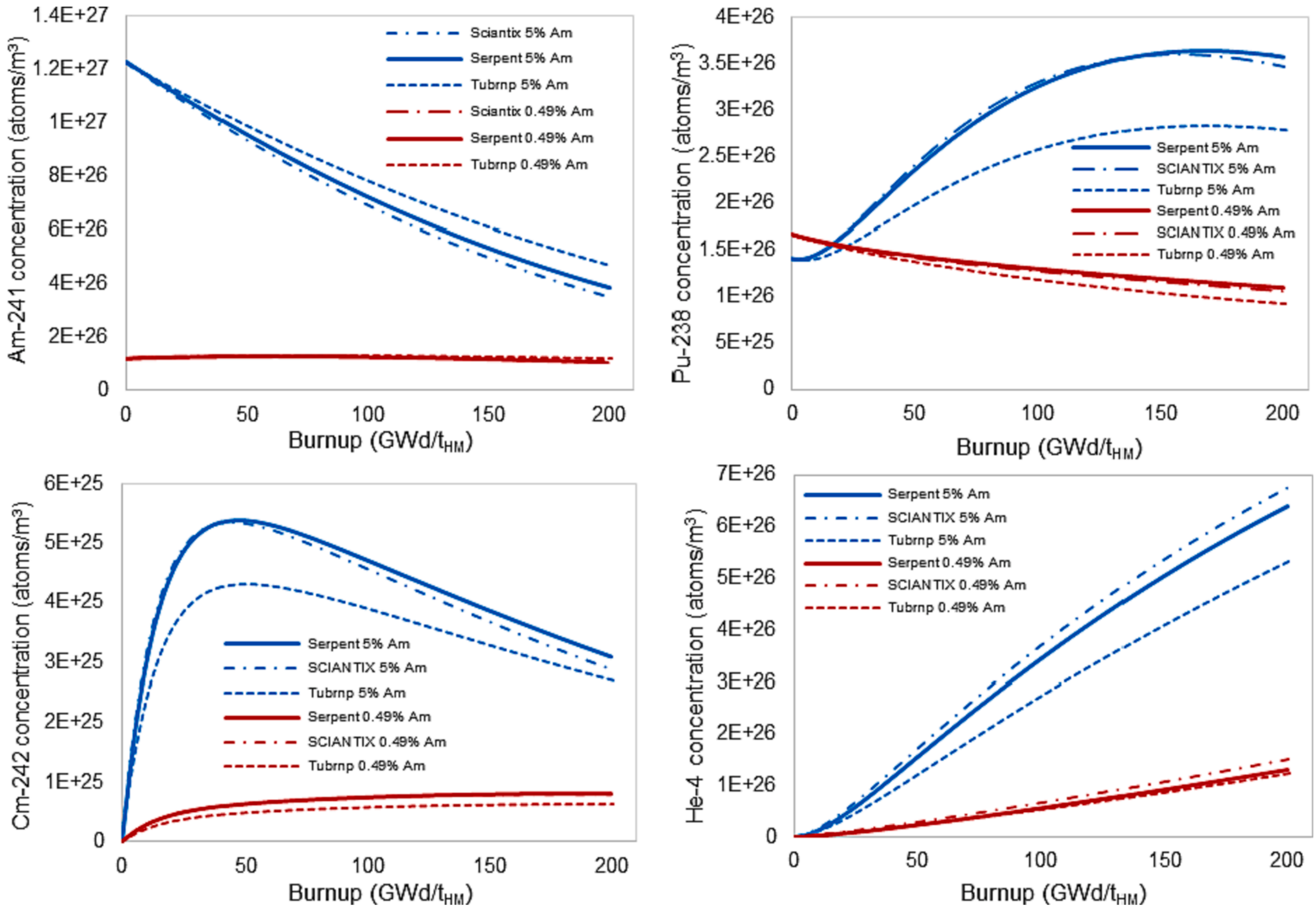

Fig. 3. Evolution of the average concentration of different nuclides at two different americium loadings, in terms of average fuel burnup, computed with TUBRNP (which is verified up to 100 GWd/t_{HM} (Cechet et al., 2021; Botazzoli et al., 2011)), the SCIANTIX burnup module extended to Am-bearing fuels and the high-fidelity SERPENT simulation (IPS irradiation conditions). For the sake of comparison, the peak discharge burnup at the end of the 90 days-long, single-cycle IPS irradiation is around 7.5 GWd/t_{HM}, while around 70 GWd/t_{HM} at the end of the 13-cycles driver irradiation.

Table 2
Ranges of irradiation data used for the SCIANTIX burnup module simulations, equipped with specific cross-section tables, to produce synthetic datasets of helium production in the MYRRHA Am-MOX fuel (MYRRHA-IPS irradiation conditions).

Variable	Range
Fission rate $\dot{F}$ (fissions m^{-3} s^{-1})	$[(1.3 \div 2.3) \cdot 10^{19}]$ [a]
Burnup (GWd t_{HM}^{-1})	$[0 \div 200]$
Am concentration (wt.%)	$[0 \div 5]$ [b]
Pu concentration (wt.%)	$[25 \div 30]$ [b]
U-235 enrichment / U (wt.%)	0.711
O/M as-fabricated (/)	1.969
Temperature (°C)	$[550 \div 1200]$ [a]

[a] The fission rate interval corresponds to the values of the IPS fuel pin at different axial positions. The temperature range covers the values experienced by the fuel during the IPS irradiation (outer temperature ÷ inner temperature) at different Am contents in the range $[0 \div 5]$ wt.%.

[b] The Am and Pu concentrations in the as-fabricated fuel are complementary each other for a total of 30 wt.%, hence 0 wt.% Am is associated to 30 wt.% Pu, while 5 wt.% Am to 25 wt.% Pu (the same holds for every intermediate value within the ranges).

Table 3
Coefficient values and associated standard errors of the proposed model for helium production in the MYRRHA Am-MOX fuel.

Coefficient	Estimate	Standard error
A	$3.387 \cdot 10^{20}$	$2.145 \cdot 10^{18}$
B	$2.717 \cdot 10^{19}$	$2.057 \cdot 10^{17}$
C	$3.528 \cdot 10^{20}$	$5.560 \cdot 10^{18}$
D	$-9.692 \cdot 10^{14}$	$1.991 \cdot 10^{13}$
E	$-2.397 \cdot 10^{14}$	$2.880 \cdot 10^{12}$
F	$2.057 \cdot 10^{15}$	$4.176 \cdot 10^{13}$
G	80.77	$1.146 \cdot 10^{-1}$
H	8.864	$3.263 \cdot 10^{-1}$
I	$-1.252 \cdot 10^{-23}$	$5.411 \cdot 10^{-26}$
J	$1.811 \cdot 10^{-23}$	$1.480 \cdot 10^{-25}$
K	$-1.970 \cdot 10^{25}$	$1.971 \cdot 10^{22}$
X	$2.218 \cdot 10^{23}$	$3.218 \cdot 10^{20}$

indicated in Section 3 (including the dedicated model for He production) and with reference to the core configuration, specifications and scenarios introduced in Section 2. The analysis focuses on two different fuel compositions of interest in terms of americium content, i.e., 0.49 wt.% which is considered as the reference for the current MYRRHA design (associated to 29.51 wt.% Pu), and 5 wt.% Am (25 wt.% Pu) as it represents the upper limit for the homogeneous Am composition. The aim is to showcase the sensitivity of pin performance figures of merit to the fuel Am content, providing indications on the possibility to load the MYRRHA fuel with more Am than in the reference as-fabricated pellets. It is worth underlining that different fuel compositions, hence different Am contents, surely correspond to different local power and fluxes at both IPS and driver irradiation positions. The assumption adopted in this work is that Am contents in the range $[0 \div 5]$ wt.% do not significantly impact on the irradiation conditions shown in Fig. 2b, referred to the

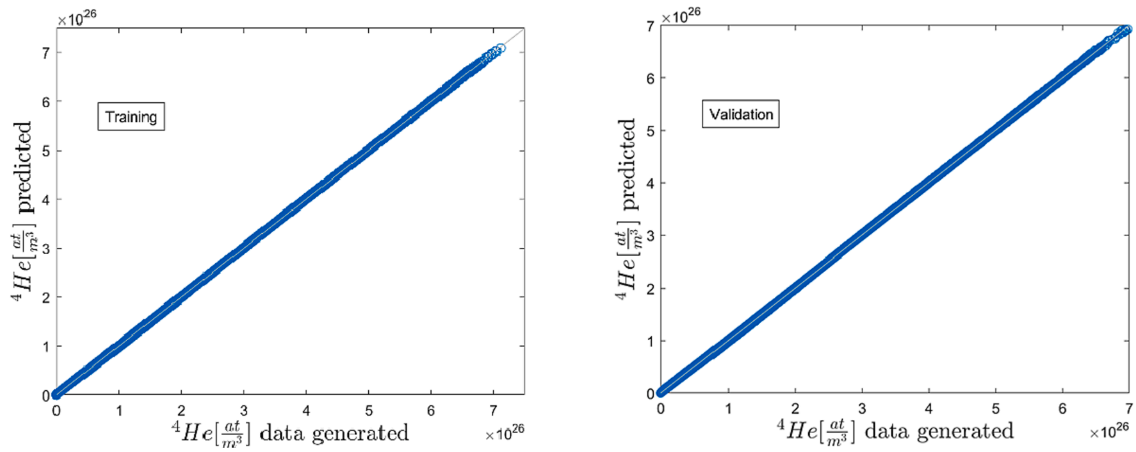

Fig. 4. Scatterplots of the predicted values of helium produced from the model proposed in this work against the values generated by the SCIANTIX burnup module (left: "training" dataset, right: independent "validation" dataset).

0.49 wt.% Am content and currently available via the PATRICIA Project (European Union's Horizon, 2020). The pin performance computations for the 5 wt.% Am fuel are thus carried out by providing the same linear power and neutron flux to TRANSURANUS//SCIANTIX.

First, simulation outcomes related to safety limits imposed on the pins in the current MYRRHA design are illustrated in Section 4.1. Then, Section 4.2 provides insights on the inert gas behaviour accounting for the impact of different Am contents, while an extended sensitivity analysis involving additional key parameters / models for the pin thermal–mechanical performance is provided in Section 4.3.

4.1. Safety figures of merit

It is fundamental to focus first on the pin safety under irradiation to ensure that relevant operational values, as provided by fuel performance code computations, never reach the design limits set for MYRRHA (Magni et al., 2022), both under IPS and driver irradiation. One design limit is set on the peak fuel temperature, whose maximum value must be lower than the conservative value of 2600°C during any reactor operative condition (Magni et al., 2022) to avoid issues related to incipient fuel melting. Fig. 5 – left shows the wide safety margin respected by MYRRHA pins loaded with Am-MOX fuel at the reference Am composition (0.49 wt.%), since the maximum fuel temperature overall reaches ~ 1450°C both during cycle 1 (when the pin maximum linear power occurs, i.e., 22.35 kW m^{-1}) and cycle 6 (when the power decreases at ~ 15 kW m^{-1} but the burnup effect on the fuel thermal conductivity plays a significant role (Magni et al., 2021)). If the IPS scenario is considered, with fresh fuel introduced in the experimental rig within the second MYRRHA core ring for a single-cycle irradiation at 16.87 kW m^{-1} (peak power), the maximum fuel temperature is limited to ~ 1110°C just after the start-up power rise, when the fuel-cladding gap is the widest and hence gap conductance the lowest.

If the 5 wt.% Am content in the as-fabricated fuel is considered, its impact on the fuel central temperature at beginning of irradiation conditions (the worst for the fuel temperature) is indicated in Fig. 5 – right. An increased Am concentration has a negative impact on the margin to fuel melting since the predicted fuel temperatures increase due to the lower thermal conductivity of Am-MOX fuel compared to U-Pu MOX (Magni et al., 2021; Magni et al., 2020). Additionally, the melting (solidus) temperature of oxide fuel is degraded by the Am content, but still higher than the conservative design limit set at 2600°C (Magni et al., 2021). For the 5 wt.% Am-MOX fuel, the maximum temperature during IPS irradiation increases up to almost 1200°C, while up to almost 1600°C at the beginning of cycle 1 of the driver irradiation. Hence, it never gets close to the limit value against fuel incipient melting issues, confirming the safety under irradiation of MYRRHA Am-MOX fuels, as

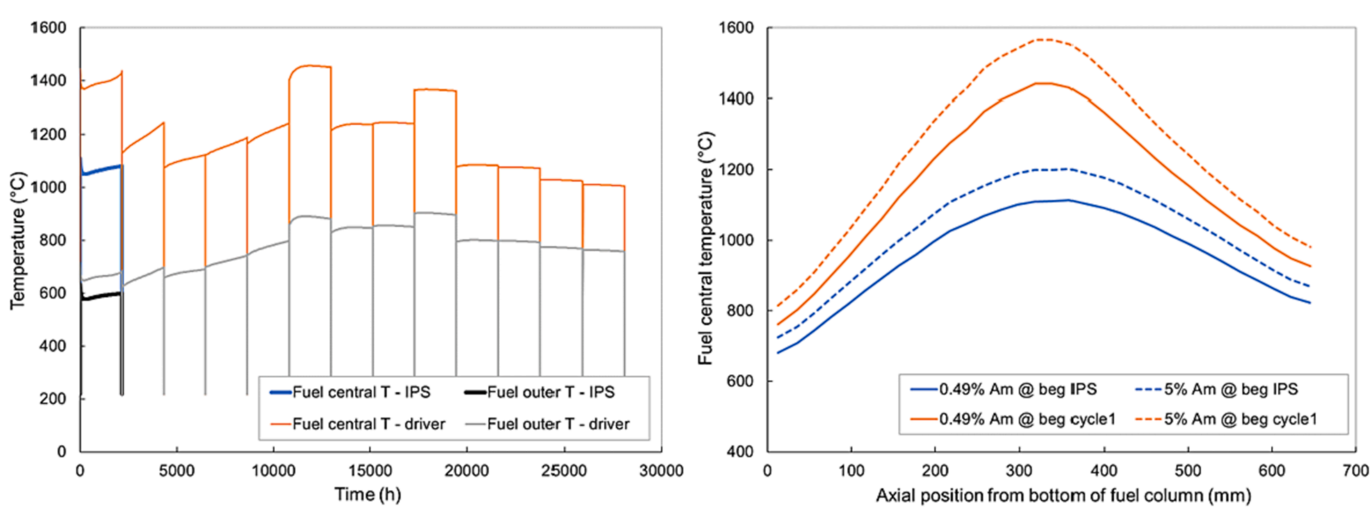

Fig. 5. Left: Evolution of the fuel central and outer temperature at the peak power node during the MYRRHA IPS and driver irradiation, for the reference Am content (0.49 wt.%). Right: Axial profiles of fuel central temperature at beginning of irradiation in the two scenarios (IPS / driver), for two different fuel Am contents (0.49 wt.% - reference and 5 wt.%).

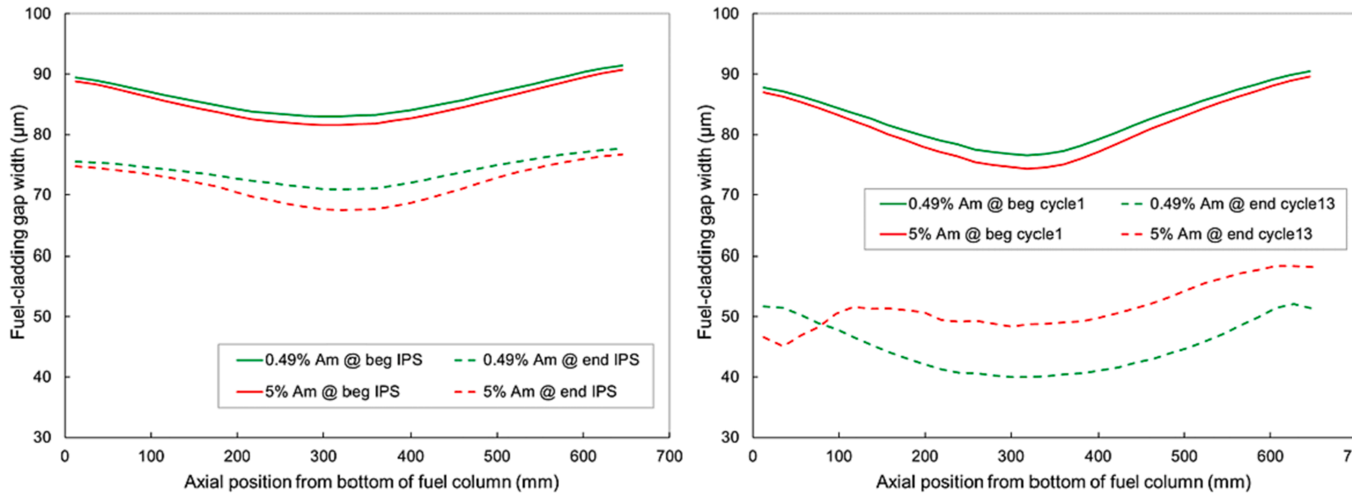

Fig. 6. Axial profiles of fuel-cladding gap width during the MYRRHA IPS irradiation (left) and driver irradiation (right), for two different fuel Am contents (0.49 wt. % - reference and 5 wt.%).

already demonstrated for U-Pu MOX fuel (Magni et al., 2022).

A second design limit concerns the plastic strain of the cladding, which is allowed up to 0.5% (Magni et al., 2022). Plasticity is avoided if the material behaviour remains within the elastic regime, i.e., the equivalent stress is below the yielding stress, which can be conservatively estimated at around 400 MPa for DIN 1.4970 under the worst MYRRHA conditions in terms of cladding temperature (Magni et al., 2022; Magni et al., 2023). No fuel-cladding mechanical interaction (gap closure) is observed from the simulations performed of Am-MOX pins under MYRRHA normal operating conditions (Fig. 6), so the only contributors to the cladding stress state are the pressure loadings on the cladding (from the gas in the gap on the inner surface, while from the coolant on the outer side), and the thermal gradient across the cladding causing thermal stresses. This results in limited values of von Mises equivalent stress in the cladding during both IPS and driver irradiations, i.e., the maximum values are ~ 53 MPa at the beginning of the IPS cycle and ~ 69 MPa at the beginning of the driver irradiation, both at the cladding inner surface and independently of the fuel Am content. These values are well below the cladding yield stress under any MYRRHA condition and consequently below the safety limit preventing the cladding plasticity. For what concerns the cladding deformations during irradiation, irradiation creep and cumulative void swelling are dominant during both IPS and especially driver irradiation, according to their modelling specific for the DIN 1.4970 cladding steel (Magni et al., 2022; Grossbeck et al., 1990; Lemehov et al., 2011). Instead, thermal creep does not play a significant role in the behaviour of the MYRRHA cladding, considering the limited temperature and temperature gradients across the cladding (at every axial location).

Fig. 6 provides the evolution of fuel-cladding gap width, which remains always open even at the end of the 13-cycles driver irradiation scenario, hence no concerns arise about fuel-cladding mechanical or chemical interaction (which can potentially result in cladding inner corrosion at gap closed). The gap progressively reduces under irradiation at every pin axial location, since driven by the differential radial deformations of fuel and cladding, caused by thermal expansion, creep and swelling, and dominated by the fuel (hotter than the cladding). The minimum gap width occurs at the end of cycle 13 just before reactor shutdown, and corresponds to 40 μm for the pin loaded with 0.49 wt.% Am-MOX. A larger final gap for the 5 wt.% Am-MOX case (Fig. 6 – right) is explained by the thermal feedback provided by a significantly higher helium production and release during driver irradiation compared to the reference fuel composition, as discussed in the following Section 4.2. The substantial amount of He in the gap has a beneficial effect on the gap conductance (compared to a gap composition dominated by gaseous fission products – Xe and Kr), hence the temperature regime in the fuel becomes slightly lower, which limits the fuel deformation mechanisms

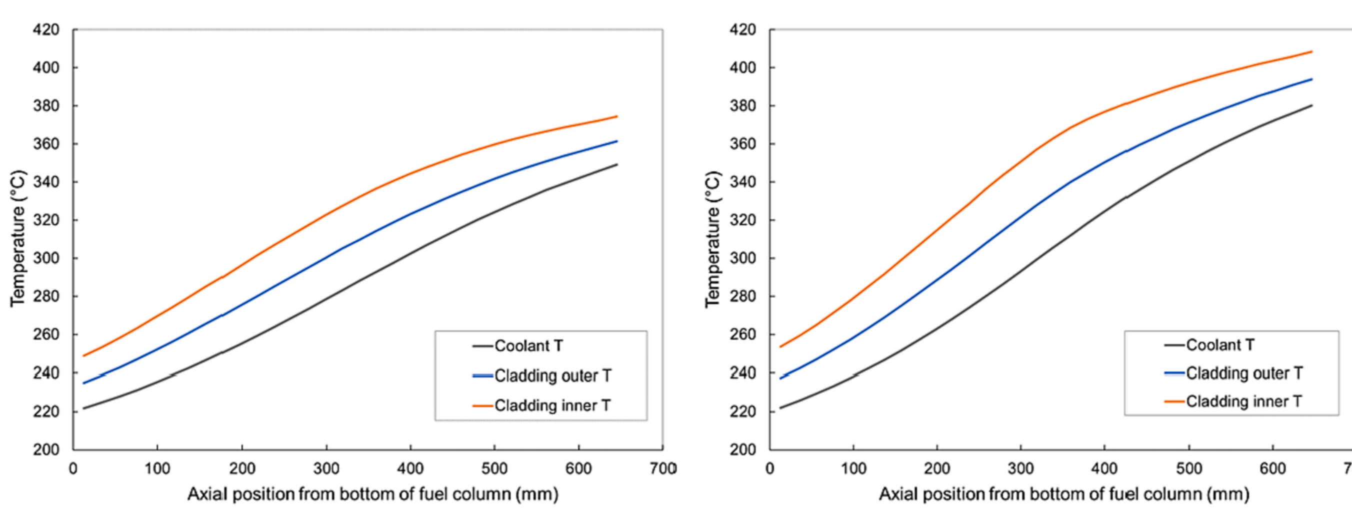

Fig. 7. Axial profiles of coolant temperature, cladding outer and inner temperatures at the end of the MYRRHA IPS irradiation cycle (left) and of the first cycle of the driver irradiation (right).

and keeps the gap wider.

Given the LBE cooling environment, a design limit is also set to prevent external cladding corrosion from the coolant, i.e., the coolant temperature must never exceed 400°C, according to a lowered (conservative) safety threshold for MYRRHA "Revision 1.8" to further increase the safety margins during normal operation. Moreover, a lower inlet coolant temperature is currently adopted (i.e., 220°C compared to 245°C in MYRRHA "Revision 1.6" (Magni et al., 2022; Magni et al., 2023; Magni et al., 2022)), together with a lower pin power (16.87 kW m^{-1} compared to 19.5 kW m^{-1}, in the second ring where the IPS is located). As shown in Fig. 7 - left, the IPS scenario is clearly within these limits along the entire pin axis (independently of the fuel Am content), with the coolant maximum (outlet) temperature ~ 350°C and the cladding outer temperature reaching ~ 362°C at the top of the fuel column after an irradiation cycle under IPS conditions. At the first cycle of the driver irradiation (within the first core ring, the closest to the central spallation channel, and hence at the highest MYRRHA power), the safety threshold is still respected with a lower margin. The coolant outlet temperature is around 380°C and the cladding outer temperature reaches ~ 395°C at the top of the pin (Fig. 7 – right), values still limiting the concerns about cladding outer corrosion [5].

4.2. Inert gas behaviour

The behaviour of inert gases (fission gases – xenon and krypton, and helium) within the fuel is relevant to the pin performance assessments since it impacts the thermal conductance of the fuel-cladding gap via gas release from the fuel, and the fuel swelling coherently. Moreover, the gas retained in the fuel contributes to the degradation of fuel thermal–mechanical properties (e.g., thermal conductivity), while the gas release determines the gap pressurization and hence the mechanical stress levels in the cladding.

The SCIANTIX grain-scale code, coupled to TRANSURANUS as an inert gas behaviour module (Zullo et al., 2023; Van Uffelen et al., 2020; Pizzocri et al., 2021; Magni et al., 2023; Pizzocri et al., 2019) (Section 3), allows a physics-based description of gas behaviour in the fuel leading to coherent evaluations of fission gas release and gaseous swelling. The coupling hence enables the evaluation of integral effects of gas-related dynamics, including helium and its production in MYRRHA Am-oxide fuels according to the specific model applicable to both the IPS and driver irradiation scenarios (Section 3.2). This surrogate model represents explicitly the He production by Am-241 and also, in an engineering way via the irradiation time variable, the contribution to He production of nuclides that are produced via neutron absorption on Am-241, e.g., Cm-242 and Pu-238 which are strong α-emitters as well. For this reason, if different axial locations along the fuel stack are considered, He production is enhanced at the peak power node where the neutron flux is maximum.

The novel model tailored for MYRRHA Am-MOX provides a higher helium source term compared to that predicted by the standard burnup module embedded in the TRANSURANUS code (TUBRNP (Botazzoli et al., 2011; European Commission, 2022)), already in the IPS irradiation case (Fig. 8 - left) with even larger differences during prolonged irradiations (driver scenario). This effect is visible especially at 5 wt.% initial Am content in the fuel, while for the reference 0.49 wt.% Am content the He production predicted by the surrogate model or by TUBRNP is similar. Nevertheless, even at 5 wt.% Am content, the integral effects on the overall pin performance during the IPS irradiation are limited, e.g., concerning the gap pressure evolution (Fig. 8 – right). Indeed, the gas diffusion phenomena and gas bubble dynamics that occur within the fuel, where inert gases are produced, are heavily influenced by the working temperatures, i.e., the higher the temperature, the faster the inert gases diffuse to the grain boundaries, where they accumulate and will eventually be released into the pin free volumes (gap, plenum) (Zullo et al., 2023). The relatively low temperatures experienced by the fuel in the current MYRRHA design (Fig. 5) and the short duration of the IPS irradiation determine that a negligible amount of inert gases is released in the gap, as indicated by the slight gap pressure increase up to less than 0.2 MPa at the end of the IPS single-cycle. Increasing the fuel Am content corresponds to a hotter fuel, hence to an enhanced diffusion of gas to the grain boundaries and consequent gas release as a result of the inter-granular gas dynamics modelled in SCIANTIX (Pizzocri et al., 2020; Zullo et al., 2023; Pastore et al., 2013).

Fig. 9 shows the production and release of inert gases during the entire driver irradiation, for the reference 0.49 wt.% fuel Am content and for the 5 wt.% Am as upper limit considered in this work. While the fission gas production is similar for the two fuel compositions under the same irradiation scenario, the helium production is significantly enhanced at 5 wt.% Am, by a factor of ~ 10 at the end of cycle 13 (by comparing the full blue lines of Fig. 9 – left and right), contributing to both fuel swelling and gap pressure increase due to a substantial helium release. Indeed, the gap pressure continuously increases up to ~ 1.6 MPa and 1.9 MPa at the end of the driver irradiation, for the 0.49 wt.% and 5 wt.% Am-MOX pins, respectively (Fig. 10). Nevertheless, as already explained in Section 4.1, even at the end of the pin life in MYRRHA, the safety limit on the cladding plasticity is not challenged, and the pin performance keeps safe from both the mechanical and thermal points of view. The latter is linked to the beneficial role played by helium release on the gap conductance (i.e., heat transfer from fuel to cladding), determining a colder fuel towards the end of driver irradiation if a bigger amount of helium is released. This reflects in lower fission gas and helium releases (in terms of percentage of the amounts of inert gas produced, dashed lines in Fig. 9) in the 5 wt.% Am-MOX case compared to the 0.49 wt.% Am-MOX case.

4.3. Sensitivity analysis on safety parameters

The fuel americium content is not the only parameter that can impact on safety-relevant figures of merit. There are multiple other factors that must be considered, some of which come with significant uncertainties. Hence, a sensitivity study is performed via a Pareto analysis (Wilkinson, 2006; Franch et al., 2016; Alkiayat, 2021) to assess how significant is the impact of the as-fabricated fuel Am loading (within the homogeneous concentration range [0 ÷ 5] wt.%) on safety-related simulation results, compared to that of other variables accounting for their intrinsic uncertainties. Uncertainty is intended here both as the modelling one, linked to the choice of different models for the same phenomenon or property, and the experimental one associated to the available measurements of a certain property or parameter. The properties / phenomena considered and the associated ranges of values or models explored are collected in Table 4. The sensitivity analysis is applied here to the single-cycle IPS irradiation, to draw first but meaningful MYRRHA-oriented indications extendable to the complete driver fuel irradiation.

As already stated in Section 4.1, the margin to fuel melting is a fundamental safety figure of merit that must comply with a design limit

[5] The oxidation of the cladding outer surface caused by the corrosive action of the LBE coolant is not considered by the simulations presented in this work, due the current lack (generally in literature and in the TRANSURANUS code) of a corrosion model specific to the couple DIN 1.4970 cladding – LBE coolant and to the MYRRHA conditions. This is recognized as a relevant future development, and a dedicated model will be implemented in TRANSURANUS and tested on MYRRHA as soon as it will be available. Additional effects impacting on the cladding temperature (and hence on the corrosion dynamics) are local hotspots due to the wire spacers, and the boundary layer associated to the LBE coolant. These are out of the modelling capabilities of the TRANSURANUS fuel performance code, but can be accounted for by an advanced modelling of the coolant sub-channel via high-fidelity thermal-hydraulics codes / capabilities, e.g., via OpenFOAM. This kind of approach is under development and would be the target of a follow-up dedicated publication.

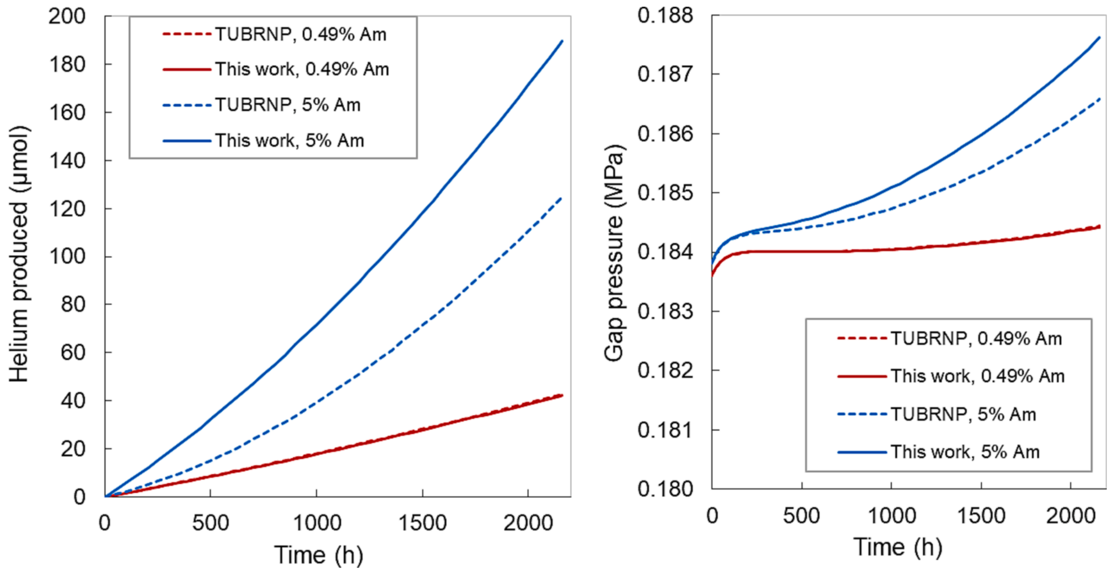

Fig. 8. Evolution of helium production during the MYRRHA IPS irradiation, for two different fuel Am contents (0.49 wt.% - reference and 5 wt.%). The results obtained with the standard TRANSURANUS code (TUBRNP burnup module) are compared with those from TRANSURANUS//SCIANTIX equipped with the He production model for MYRRHA Am-MOX fuels developed in this work.

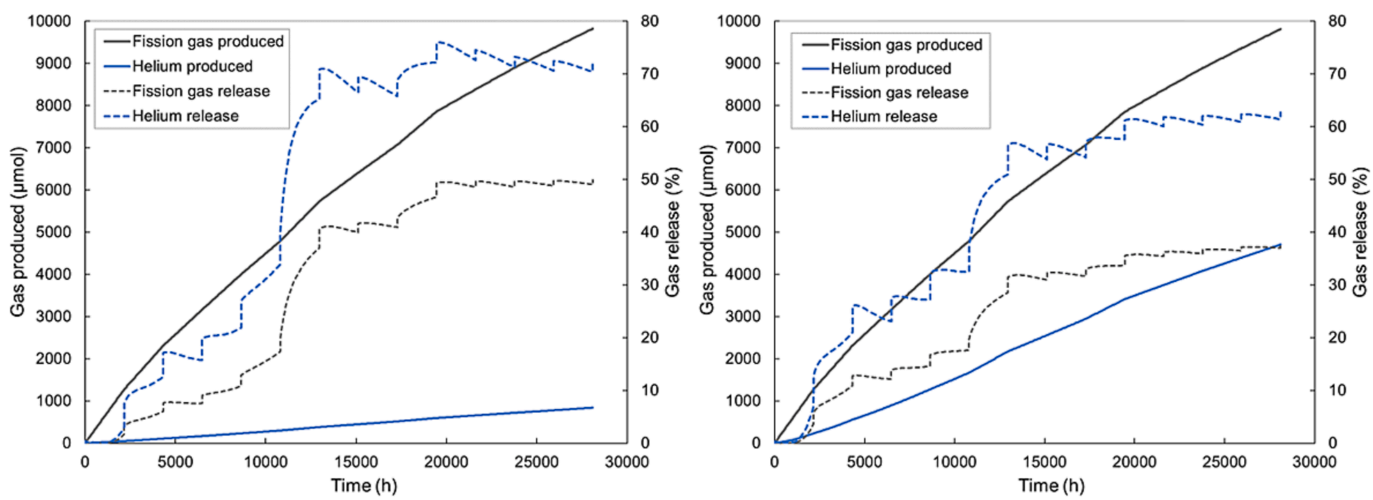

Fig. 9. Evolution of the production and release of inert gases during the MYRRHA driver irradiation of Am-MOX fuels, for two different Am contents: 0.49 wt.% - reference (left) and 5 wt.% (right).

during any operative irradiation condition (both normal operation and transient scenarios). As shown by Fig. 11, it is mostly influenced by the choice between different models of gap conductance (Lassmann and Hohlefeld, 1987; Charles and Bruet, 1984), while the impact of different initial Am concentrations between 0 and 5 wt.% in the fuel [6] is lower and comparable to that attributed to the uncertainty on fuel thermal conductivity (Magni et al., 2021) or to the activation of the coupling with SCIANTIX for the physics-based inert gas behaviour (Zullo et al., 2023) (already applied to fast reactor cases (Luzzi et al., 2023; Magni et al., 2022), and to MYRRHA in (Magni et al., 2023).

Fig. 12 - left shows that the fuel Am content in the range [0 ÷ 5] wt.% has a limited (indirect) effect on the gap pressure and hence on the cladding stress levels when the gap is open, via the Am role on the production of helium, whose dynamics and release is considered by SCIANTIX, and on the fuel thermal conductivity. Specifically, the gas release in the fuel-cladding gap is slightly impacted by the fuel Am loading since it influences the fuel temperature regime driving the gas diffusion within the fuel matrix. Nevertheless, the Am effect on the gap pressure is obscured by other parameters and phenomena strongly acting on the fuel temperature, i.e., the thermal boundary condition provided by the external LBE coolant (in terms of both inlet temperature and heat transfer coefficient between cladding and coolant) and the gap conductance. Section 4.2 already illustrated the limited over-

[6] The fuel Am content impacts on safety-relevant figures of merit not just via the fuel composition (different enrichment levels), but also via its effect on fuel properties, e.g., thermal properties such as the thermal conductivity and melting temperature, fundamental for the margin to fuel melting. A deep analysis of their dependencies on the Am content is presented in (Magni et al., 2021) (providing the recent models for thermal properties of Am-MOX employed in this work - Section 3.1) and in previous literature works cited therein. Americium is identified as a relevant contributor to the degradation of the fuel thermal conductivity, together with the temperature and irradiation effects, while its impact on the melting temperature is small compared to that associated to deviation from fuel stoichiometry (oxygen-to-metal ratio) and burnup.

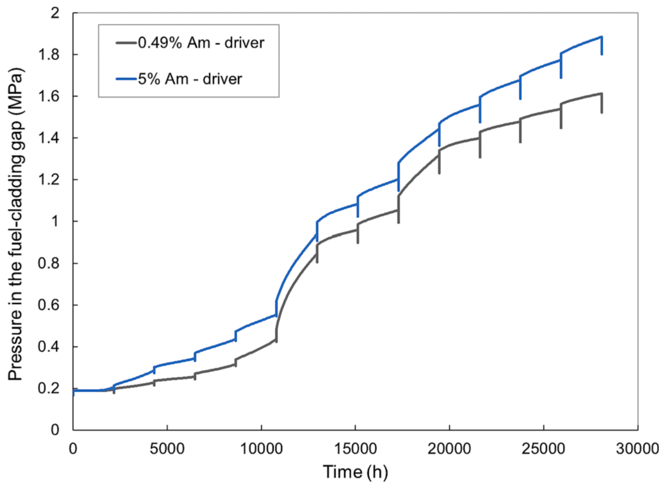

Fig. 10. Evolution of the pressure in the fuel-cladding gap during the MYRRHA driver irradiation of Am-MOX fuels, for two different Am contents: 0.49 wt.% - reference and 5 wt.%.

Table 4
List of parameters / phenomena involved in the sensitivity analysis to assess their impact on safety figures of merit, by considering the associated modelling / experimental / design uncertainties [7].

Parameter / Phenomenon	Models / uncertainty
Fuel Am content	[0 ÷ 5] wt.%
Gap conductance	Models available in TRANSURANUS (Lassmann and Hohlefeld, 1987; Charles and Bruet, 1984) ± 30% (Lassmann and Hohlefeld, 1987; Pastore, 2012; Luzzi et al., 2014)
Fuel densification	FBR models available in TRANSURANUS (Dienst et al., 1979; Clement, 1977)
Fuel thermal conductivity	± 20% (Magni et al., 2021; Magni et al., 2020)
Linear power	± 5%
LBE Nusselt number	± 25% (Mikityuk, 2009; Di Gennaro et al., 2023)
LBE inlet temperature	± 10°C
SCIANTIX coupling	On / off

[7] The parameters / phenomena involved in the sensitivity analysis are relevant for their (direct) impact on the compliance with the safety limits currently set for MYRRHA. Specifically, the linear power, the LBE Nusselt number and LBE inlet temperature determine the cladding temperature (corrosion), while the fuel Am content, the gap conductance modelling, fuel densification and thermal conductivity impact the fuel central temperature (margin to fuel melting). The treatment of the inert gas behaviour and release, linked to the coupling of TRANSURANUS with SCIANTIX, provides the mechanical loading on the cladding (open gap condition) and consequently the cladding stress state. The uncertainties involved are assumed based on literature indications (referenced in Table 4) or are suggested by the MYRRHA design team (within the PATRICIA Project) for what concerns the fuel Am content, the related pin linear power, and the inlet temperature of the LBE coolant.

pressurization of the 5 wt.% Am-MOX pin compared to the 0.49 wt.% one, under both IPS and driver irradiation scenarios.

Moreover, Fig. 12 – right confirms that the fuel Am content has no effect on the inner / outer cladding temperatures (crucial to prevent cladding corrosion), as anticipated in Section 4.1. This is expected since the cladding thermal performance is dominated by the inlet temperature of the coolant and by its heat transfer coefficient with the cladding. Thus, it can be concluded that the only safety figure of merit that is significantly influenced by the initial americium content in the fuel is the margin to fuel melting, mainly because of the effect that americium has on both thermal conductivity and melting temperature of oxide nuclear fuels (Magni et al., 2021), and on the gap conductance via helium release as discussed in Section 4.2.

5. Conclusions and further developments

The current design of the MYRRHA sub-critical reactor ("Revision 1.8") is equipped to house Am-MOX experimental fuel pins for irradiation experiments and safety studies, and the pin performance analyses performed in this work confirm that the use of such fuel as a MYRRHA driver fuel is a feasible and safe option since the pin behaviour under irradiation keeps widely within the safety design limits. This work hence supports an extended range of fuel options for MYRRHA, after the safety under both normal operation and over-power transient conditions of U-Pu MOX fuel already demonstrated in previous works from the same authors (Magni et al., 2022; Magni et al., 2023).

The calculations performed in the present analysis focused on two irradiation scenarios, i.e., a single power cycle within the IPS position in the second ring of the MYRRHA core, and the use of the same Am-MOX as driver fuel along the 13 cycles of MYRRHA normal operation, along five core rings up to the final assembly discharge. Besides the reference Am content in the as-fabricated fuel (currently set at 0.49 wt.%), the possibility of increasing the Am loading to 5 wt.% (limit of the homogeneous Am recycling strategy) is accounted for, investigating the associated effects and consequences. A relevant one is the enhanced helium production within the fuel, now coherently estimated via a surrogate model specific for the MYRRHA scenarios considered. The model is derived from synthetic datasets of helium produced under different fuel compositions and irradiation conditions corresponding to the current MYRRHA design, generated by a version of the SCIANTIX burnup module capabilities (Cechet et al., 2021) extended to Am-bearing oxide fuels (Pizzocri et al., 2023). The model enables to represent explicitly and coherently, in an engineering way suitable for coupling with integral fuel performance codes, the impact of increasing Am contents on the helium production and behaviour, reflecting on pin-level quantities as the gas release, gap pressurization and fuel temperature in turn. However, these effects have a negligible impact on the MYRRHA safety criteria considered in this work (margin to fuel melting, allowed cladding plasticity and prevented outer corrosion of the cladding).

Additional analyses herein included focus on the sensitivity of safety figures of merit to the fuel Am content compared to the impact of other relevant parameters and phenomena driving the pin performance (e.g., gap conductance, linear power, coolant conditions). The latter study indicates the dominant role of the gap conductance (uncertainty and modelling) and of the thermal boundary conditions provided by the coolant, while the Am effect proves to be secondary and indirect (an increasing Am loading slightly reduces the margin to fuel melting, which keeps anyway large, and contributes to the gap pressurization via the increasing helium production).

Further developments emerging from this work can be identified. First, the SERPENT-informed SCIANTIX methodology for a surrogate model for helium production, herein applied to MYRRHA Am-MOX fuels, is suitable to be applied to other fuel materials (especially Am-bearing, where He production is significant) and irradiation conditions. Moreover, the current model is built on artificially generated datasets that only consider constant irradiation conditions, i.e., normal operation conditions both in the IPS or during each of the 13 driver cycles, without accounting for transient scenarios. The model should be hence extended to be properly applicable to any kind of potential irradiation conditions in MYRRHA (or other reactor concepts), including power ramps (of start-up / shutdown, and also operational and accidental transients). Moreover, a refinement of the surrogate model to accurately follow the He production during storage or annealing phases can be achieved via the generation of specific synthetic datasets.

From the modelling point of view, the present work shows the importance of focusing follow-up efforts in better assessing the gap conductance under fast reactor conditions. This is identified from the Pareto sensitivity analyses performed as a strong responsible for the fuel pin performance and hence for the associated design evaluations.

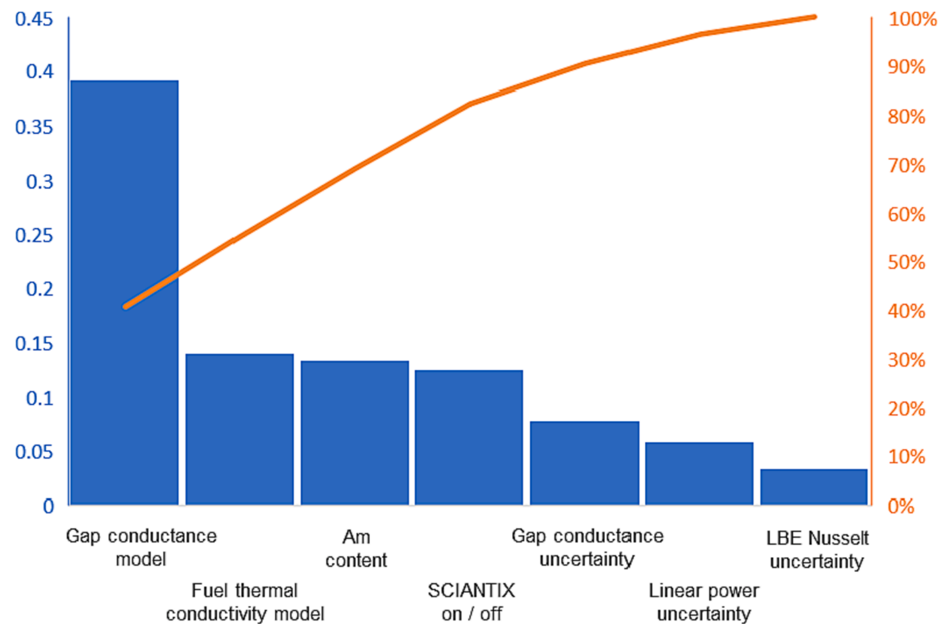

Fig. 11. Pareto diagram showing the sensitivity of the margin to fuel melting to different parameters or phenomena, according to their current modelling in TRANSURANUS//SCIANTIX and to the associated uncertainties.

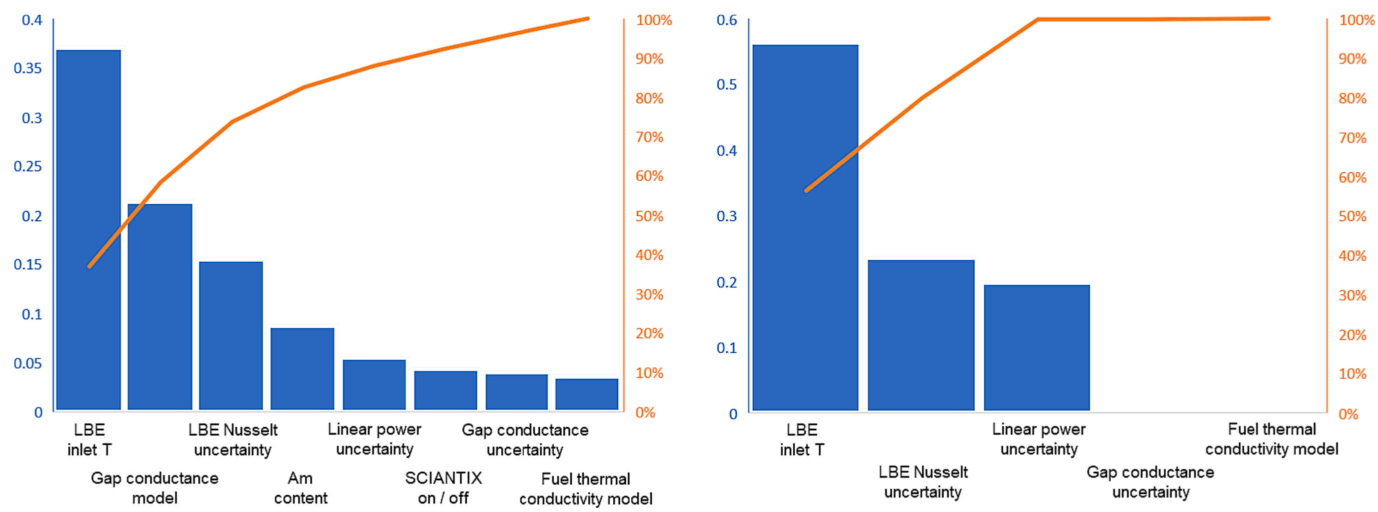

Fig. 12. Pareto diagrams showing the sensitivity of the gap pressure (left) and cladding outer temperature (right) to different parameters or phenomena, according to their current modelling in TRANSURANUS//SCIANTIX and to the associated uncertainties.

Moreover, a general advancement relevant for advanced fuel performance code simulations and safety assessments under LBE (and lead) cooling environments consists in the development and use of a reliable and dedicated model for the corrosion of the cladding outer surface (specific for a certain cladding-coolant couple, besides depending on the surface temperature and on the oxygen control strategy).

The consideration of transient and accidental irradiation conditions is of high interest also from the pin performance and safety points of view, on the basis of the simulation results on Am-MOX fuels under MYRRHA normal operation herein achieved. Additionally, while the present work is limited to the "homogeneous" Am content range [0 ÷ 5] wt.% ((U, Pu, Am)O_{2-x} driver fuel design), additional fuel performance analyses (supported by dedicated neutronic modelling and boundary conditions) can be applied to fuel compositions covering the "heterogeneous" americium loading, i.e., a (U, Am)O_{2-x} mixed-oxide fuel concept with Am enrichments up to 15 wt.% (D'Agata et al., 2017). Since such a composition (which uses natural U) does not contain significant quantities of fissile materials, it is of interest as blanket fuel to be loaded in the outer power channels of the MYRRHA reactor core for transmutation purposes.

CRediT authorship contribution statement

L. Luzzi: Conceptualization, Funding acquisition, Project administration, Resources, Supervision, Visualization, Writing – original draft, Writing – review & editing. **A. Magni:** Conceptualization, Data curation, Investigation, Methodology, Software, Validation, Visualization, Writing – original draft, Writing – review & editing. **S. Billiet:** Conceptualization, Data curation, Writing – review & editing. **M. Di Gennaro:** Investigation, Methodology, Software, Validation, Writing – review & editing. **G. Leinders:** Conceptualization, Writing – review & editing. **L.G. Mariano:** Investigation, Software, Validation, Writing – review & editing. **D. Pizzocri:** Conceptualization, Methodology, Writing – review & editing. **M. Zanetti:** Writing – review & editing. **G. Zullo:**

Writing – review & editing.

Declaration of competing interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Data availability

Data will be made available on request.

Acknowledgments

This work has received funding from the Euratom research and training programme 2019-2020 under grant agreement No 945077 (PATRICIA Project).

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