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D6.2 Transient simulation of the in-pile performance of a MA-bearing fuel pin

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DOCUMENT SUMMARY

This document reports the performance evaluation of the hottest americium-bearing fuel pin irradiated in the in-pile test sections (IPS) sub-assembly of the MYRRHA research reactor (design revision 1.8) during the primary over-power transient scenario selected within the PATRICIA Project (Milestone MS5), namely the Beam Power Jump (BPJ). The simulations are performed with the TRANSURANUS thermo-mechanical fuel performance code coupled with the SCIANTIX physics-based module for inert gas behaviour, incorporating the propaedeutic advancements for transient simulations achieved within the scope of PATRICIA Work Package 5. These advancements include models for Am-fuel thermal properties (thermal conductivity, melting temperature, specific heat) developed in Task 5.1, a dedicated surrogate model for the helium production rate during MYRRHA irradiation accounting for the relevant contribution of the fuel americium enrichment in Task 5.2 and the multi-physics environment implemented in Task 5.4. The safety evaluation of the IPS fuel pin is performed on critical figures of merit (bulk temperature, fuel central temperature, gap width and equivalent von Mises stress) and reveals a good margin to the design limits.

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1 Introduction

This document aims at evaluating the safety performance of the americium-bearing fuel pins irradiated in the in-pile test sections (IPS) sub-assembly of the MYRRHA reactor during the Beam Power Jump (BPJ) over-power transient scenario. This study completes the assessment of the behaviour of the americium-bearing fuel pins under normal operation (P. Van Uffelen et al., 2024-1).

The IPS sub-assemblies are hosted in the sub-critical (acceleration-driven) configuration (70 MWth thermal power) of the MYRRHA reactor (core design version 1.8, Fig. 1) and dedicated to experimental irradiation under fast neutron fluxes of innovative americium-bearing fuel pins (core design version 1.8) as driver-type U-Pu-Am mixed oxide (Am-MOX) fuel (with americium concentration of 0.49 wt.% and 5 wt.%) or blanket-type U-Am-O fuel (with americium concentration of 10 wt.% and 20 wt.%). Each one of the IPS sub-assemblies consists of 127 wire-wrapped cylindrical fuel pins encased within the DIN 1.4970 cladding (15-15Ti stainless steel family). In Table 1 are reported the complete details about the MYRRHA fuel pin characteristics.

The specific accidental scenario considered, the BPJ one, is of particular concern for the accelerator-driven system (ADS) sub-critical configuration of MYRRHA (Magni et al., 2023) and represents the primary transient selected within the PATRICIA Project, as detailed in the Milestone MS5 (Lainet et al., 2022).

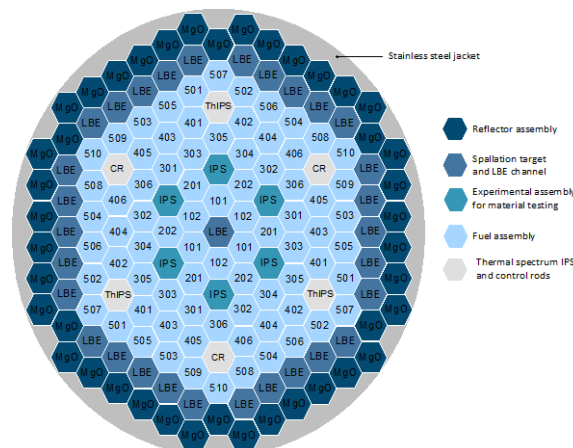


Figure 1: Scheme of the MYRRHA sub-critical core layout at beginning of cycle (BOC) according to the current design "Revision 1.8" (Fiorito et al., 2021).

The thermo-mechanical simulations are here performed with the TRANSURANUS 1.5D fuel performance code (version v1m4j22) (Lassmann, 1992; Magni et al., 2020) coupled with the grain-scale code SCIENTIX, originally developed by (Pizzocri et al., 2020) and recently advanced to the new (2.0) version (Zullo et al., 2023) developed in the framework of PATRICIA Task 5.2 (Magni et al., 2023), devoted to physics-based calculations of inert gas behaviour within the fuel matrix. The TRANSURANUS//SCIENTIX suite uses the linear heat rate and fast neutron flux, provided by SCK CEN and obtained by means of MCNP neutronic analysis of the sub-critical MYRRHA core considering the reference Am-MOX composition (0.49 wt.% americium and 29.51 wt.% plutonium) and applied here also for fuel with higher, up to 5 wt.%, americium content¹. More information about the modelling set-

¹ Although the irradiation conditions provided by SCK CEN refer to the 0.49 wt% Am content, the assumption adopted in this work is that the Am contents in the range 0.49 – 5 wt.% have a negligible impact on the MYRRHA irradiation conditions.

up employed for the performance analyses of Am-MOX fuels under MYRRHA irradiation scenarios is provided in Deliverable D5.1 (Van Uffelen et al., 2024-2) and D5.4 (Perez et al., 2023).

In the framework of the PATRICIA Work Package 5, propaedeutic developments were carried out to better simulate the BPJ over-power transient scenario, summarized in Section 3, and are here considered to perform the MYRRHA fuel pin performance analysis.

The deliverable is structured as follows. Section 2 is dedicated to describing the MYRRHA-IPS irradiation history and the specification of the BPJ transient scenario analysed. In Section 3 are presented the propaedeutic developments performed in in the framework of Work Package 5 to better simulate the BPJ over-power transient. The results are provided in Section 4 for both the OpenFOAM code results and the fuel pin performance results with particular emphasis on those figures of merit for which a safety limit has been established (Luzzi et al., 2024). Conclusions and suggested further developments arising from this work are provided in Section 5.

Table 1. Geometry specifications of the MYRRHA “Revision 1.8” core and as-fabricated MYRRHA fuel pin.

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 active 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.%
As-fabricated porosity	5%
Fuel grain size	10 µm
Fuel oxygen-to-metal ratio	1.969
Fuel density	95% TD

2 MYRRHA in-pile test section irradiation scenario

According to the MYRRHA operative schedule, the single-cycle IPS irradiation consists of 90 days of full power operation preceded by 30 minutes of reactor reaching nominal power and followed by 30 minutes of shutdown (Figure 2 - left). At any moment of the normal reactor operation in sub-critical mode, the BPJ over-power transient could eventually occur, triggered by an over-current of the external proton accelerator coupled with the MYRRHA sub-critical core. The tolerated (by design) accelerator over-current corresponds to sudden increase by (maximum) 70% of the proton source. Consequently, the power of the MYRRHA ADS core is increased by 70% in an estimated time of 1 millisecond and it is also assumed that the power in the IPS increases by the same percentage. Then, the high neutron flux signal prolonging for more than 3 seconds triggers the accelerator and reactor shut-off in 3 seconds (Figure 2 - right).

The neutron flux and the linear power that a fuel pin experiences in the IPS are assumed as the average between the two surrounding, second-ring driver positions, i.e., number 201 and 202. In particular, the values considered in this work correspond to a hypothetical hottest Am-MOX pin identified as the pin placed in the harshest position (in terms of boundary conditions and temperature) within the fuel assembly. This assumption has been made on purpose to consider the worst irradiation conditions and therefore draw conservative conclusions.

The maximum (at the axial peak power node and for the hottest pin) linear heat rate and fast neutron flux history during the MYRRHA-IPS normal irradiation and the BPJ transient are reported in Figure 2, based on data provided by SCK CEN. The BPJ transient has the potential to occur at any moment during normal reactor operation – at beginning, during or at the end of the single-cycle IPS normal irradiation. In this work, the BPJ scenarios is assumed to take place at the beginning, being the worst case for the fuel central temperature and von Mises stress in the cladding, and at the end, being the worst case for the gap width, of the IPS normal irradiation. These choices of the times of relevance for introducing the BPJ transient are supported by the assessment of the fuel pin behaviour in normal operation (P. Van Uffelen et al., 2024-1). The maximum injected specific energy in the fuel during the BPJ transient corresponds to 160 J/g.

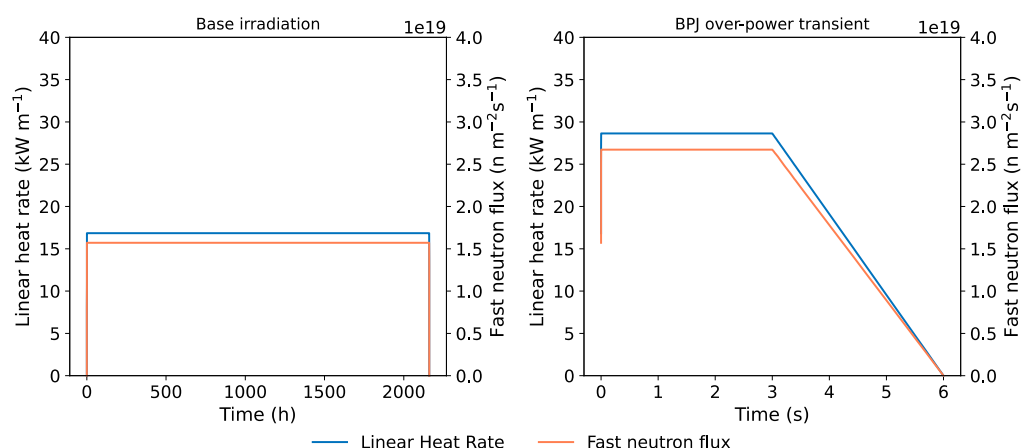


Figure 2: Irradiation conditions of the hottest IPS fuel pin at the peak power node (ppn) during the normal operation (left) and the BPJ over-power transient (right).

3 Developments propaedeutic to the transient simulations

This section outlines all the advancements achieved within the scope of Work Package 5, issued from Task 5.1 (Van Uffelen et al., 2024-2), Task 5.2 (Magni et al., 2023) and Task 5.4 (Perez et al., 2023), propaedeutic to better simulate the over-power transient scenario (BPJ) of concern for the sub-critical core configuration of MYRRHA.

3.1 Thermal properties for Am-bearing fuels

In the framework of PATRICIA Task 5.1 (Van Uffelen et al., 2024-2), the TRANSURANUS code has been equipped with updated models for the thermal properties (melting temperature, specific heat and thermal conductivity) of Am-bearing mixed oxide fuels covering the ranges associated to the MYRRHA irradiation in terms of temperature (relevant especially for the over-power scenario), americium and plutonium contents, deviation from stoichiometry, porosity and burnup. In Section 4.2 we briefly discuss the impact of incorporating the updated models for the specific heat and the thermal conductivity of Am-MOX on the fuel central temperature, in comparison to utilizing the state-of-the-art models (Chawla et al., 1981; Magni et al., 2021).

3.2 Helium production in the SCIANTIX grain-scale code

Due to the significant rise in helium production rates within the Am-bearing fuel matrix and its pivotal role in fuel pin design and performance, identified as a crucial life-limiting factor (Luzzi et al., 2018), the current release of SCIANTIX (version 2.0) (Zullo et al., 2023) has been incorporated with dedicated physics-based models for helium behaviour as part of the PATRICIA Task 5.2 (Magni et al., 2023). Emphasis was placed on developing a dedicated surrogate model for helium production in Am-bearing fuels designed for MYRRHA IPS irradiation conditions and fuel compositions (up to concentration of 5 wt.% of Am) (Luzzi et al., 2024). This model is built on synthetic datasets of produced helium generated via the burnup module capabilities of SCIANTIX itself (Cechet et al., 2021; Pizzocri et al., 2023), fed by neutronic data (cross-sections, fission yields and decay constants) derived from a SERPENT modelling and simulation of the MYRRHA IPS fuel pin (Figure 3).

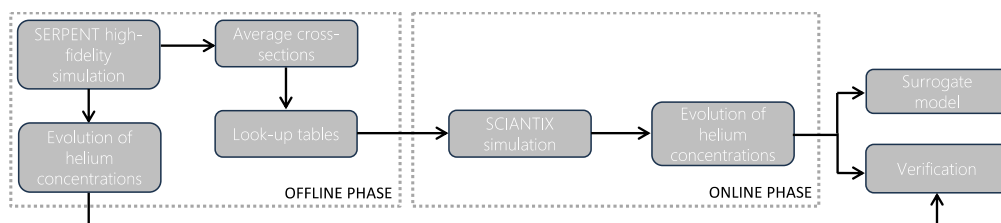


Figure 3: Schematic representation of the workflow followed in the SERPENT and the SCIANTIX code.

3.3 Fuel performance analysis in the BELLA code

Within the context of PATRICIA Task 5.4 (Perez et al., 2023), KTH enhanced the BELLA (Bortot's Elegant Liquid LFR Analysis tool) code (Bortot et al., 2015). Originally a multi-physics tool capable of integrating fluid dynamics and neutronics analysis, KTH incorporated it with a thermo-mechanical module. This module considers feedback with the thermal module, enabling the calculation of temperature profiles and the governing thermal-elastic equations. Consequently, stress, displacement, and gap width in the MYRRHA IPS fuel pin can be accurately determined, as shown in Deliverable 5.4 (Perez et al., 2023).

3.4 Neutronic and thermo-hydraulic modelling in the SIMMER code

In the framework of PATRICIA Task 5.4 (Perez et al., 2023), KIT conducted the simulation of the MYRRHA sub-critical core using the multi-physics SIMMER-III code (Kondo et al., 1992). This code integrates fluid dynamics and neutronics analysis, enabling a deeper understanding of feedback

coefficients. The analysis was performed in both normal and transient conditions and considering the MYRRHA core loaded with 5 wt.% of americium in the IPS sub-assembly, leading to more accurate power distributions and enhancing reliability.

In Figure 4 is reported the comparison between the linear heat rate and fast neutron flux (neutron energy > 0.1 MeV) at the axial peak power node of the hottest IPS fuel pin provided by SCK CEN, holding for the reference Am-MOX composition (0.49 wt.% americium and 29.51 wt.% plutonium), and by the KIT, which considers 5 wt.% Am content and 25 wt.% Pu content in the IPS position. The lower linear heat rate and fast neutron flux predicted by KIT with 5 wt.% of americium content is due to the lower fission cross section of Am241 with respect to Pu239 (Rummana et al., 2022). Moreover, during the BPJ transient, KIT predicts an increase of 60% in the IPS power which corresponds to 10% lower with respect to the assumption made with the SCK CEN data. The almost immediate power variation of about 60% is less than the beam variation of 70% because the power components related to decay heat and delayed neutron source are not in equilibrium with the prompt fission one just after the beam jump. The reactivity feedbacks come into play later.

During the BPJ transient, the maximum injected specific energy into the fuel considering the data supplied by KIT corresponds to 130 J/g.

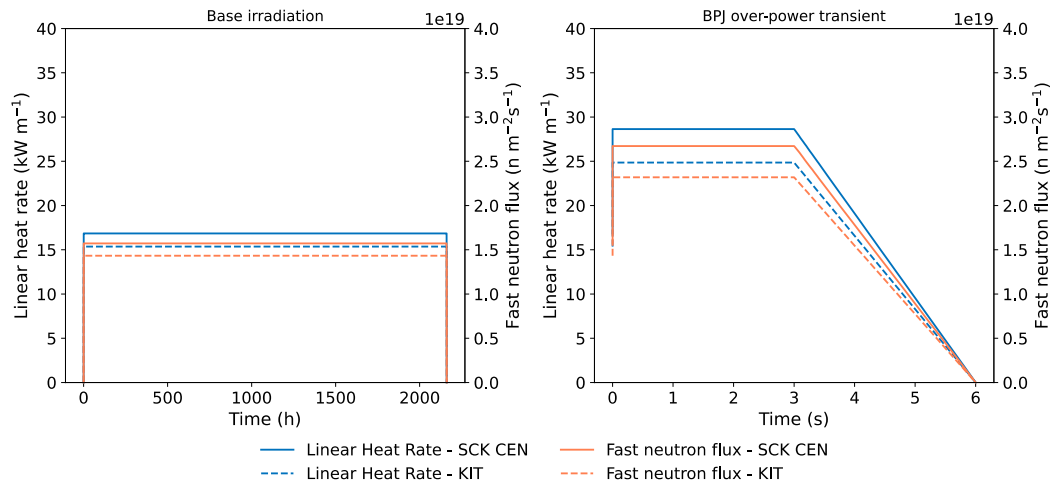


Figure 4-left: Irradiation conditions of the hottest IPS fuel pin at the peak power node, provided by SCK CEN (full line) and by KIT (dashed line). Figure 4-right: Irradiation conditions during the BPJ over-power transient of the hottest IPS fuel pin at the peak power node, provided by SCK CEN (full line) and by KIT (dashed line).

3.5 Thermo-hydraulic modelling in the OpenFOAM code

In the framework of PATRICIA Task 5.4 (Perez et al., 2023), POLIMI conducted high-fidelity computational fluid dynamics simulations of the MYRRHA-IPS interior sub-channel (see Figure 5) executed using the open-source software OpenFOAM version 6 (*OpenFOAM Version 6.0*, 2018). The solver adopted corresponds to the *buoyantBoussinesqPimpleFoam* available in the OpenFOAM version 6, appropriately modified to accurately capture the LBE-coolant flow conditions in the MYRRHA-IPS sub-channel (OECD/NEA Handbook, 2015). The dynamics of the LBE flow has been solved using the standard incompressible Navier-Stokes equations coupled with the energy equation under the Boussinesq approximation for density. Turbulence is modelled using the linear eddy viscosity model $k - \omega$ SST (Menter, 1994), employed in the low Reynolds formulation. The heat transfer is modelled adopting the gradient diffusion assumption (Shams & De Santis, 2019) and considering the local model proposed by Kays in (Kays, 1994) for the turbulent Prandtl number.

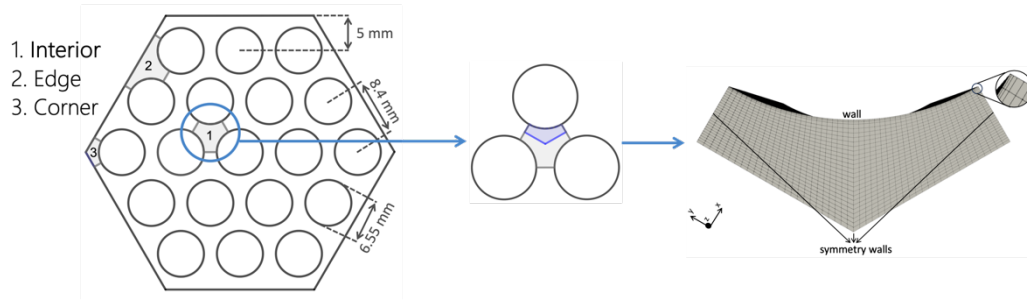


Figure 5: Illustration of the IPS sub-assembly and the close-up view of the interior sub-channel with the highlighted computational domain in blue (left and center). On the right, depiction of the coolant domain mesh.

The fuel power is prescribed as a fixed input at the wall (cladding outer surface) based on the linear heat rate data provided by SCK CEN and by KIT (Section 2.4 – Figure 3) and it follows the dynamics governed by the specific heat of the fuel obtained from the TRANSURANUS simulation described in Section 1. The main outcomes of the thermal-hydraulic calculation of the IPS interior sub-channel performed in OpenFOAM are reported in Section 4.1.

4 Results and discussion

In this section, we present and discuss the outcomes of the assessment of the MYRRHA IPS Am-MOX fuel pin performance under the Beam Power Jump transient conditions. These results are issued from calculations performed as outlined in Sections 1 and 3, and are contextualized within the core configuration, specifications, and scenarios introduced in Section 2. Section 4.1 reports the thermo-hydraulic results obtained via simulation performed in OpenFOAM of the MYRRHA interior sub-channel. Among these results, pressure drops and cladding outer temperature are utilized to inform the TRANSURANUS thermo-mechanical pin performance simulations, the findings of which are delineated in Section 4.2.

The fuel pin performance analysis in Section 4.2 focuses on two distinct fuel compositions, with varying levels of americium content: 0.49 wt.%, which is considered as the reference for the current MYRRHA design (comprising 29.5 wt.% Pu), and 5 wt.% Am (25 wt.% Pu), representing the upper limit for homogeneous Am composition. The objective is to analyse the sensitivity of pin performance figures of merit to the fuel Am content, offering insights into the feasibility of loading MYRRHA fuel with a higher Am content than that of the reference as-fabricated pellets. The irradiation conditions considered encompass those provided by SCK CEN, adopted for both fuel compositions, as well as those provided by KIT, applied specifically to the 5 wt.% Am case.

4.1 OpenFOAM results

The temperature profiles depicted here solely pertain to the linear heat rate and fast neutron flux values supplied by SCK CEN (Figure 2 and 4). The temperature profiles associated with KIT data (Figure 4) are identical, albeit rescaled.

In Figure 6 – left, the coolant temperature is illustrated at three different times: the blue line represents the coolant temperature before the occurrence of the BPJ, indicating the steady-state temperature reached during the normal operation; the orange line depicts the coolant temperature three seconds after the transient event, just before the reactor shutdown; finally, the red line denotes the maximum temperature the coolant would reach if the reactor would remain operational three seconds after the event without shutdown. The prescribed safety limit of 400°C for the coolant temperature during the normal MYRRHA operation is strictly adhered to, thus effectively preventing corrosion of the outer cladding. This limit is exceeded only during the transient (orange line), but the duration of the transient is so short that is not a problem.

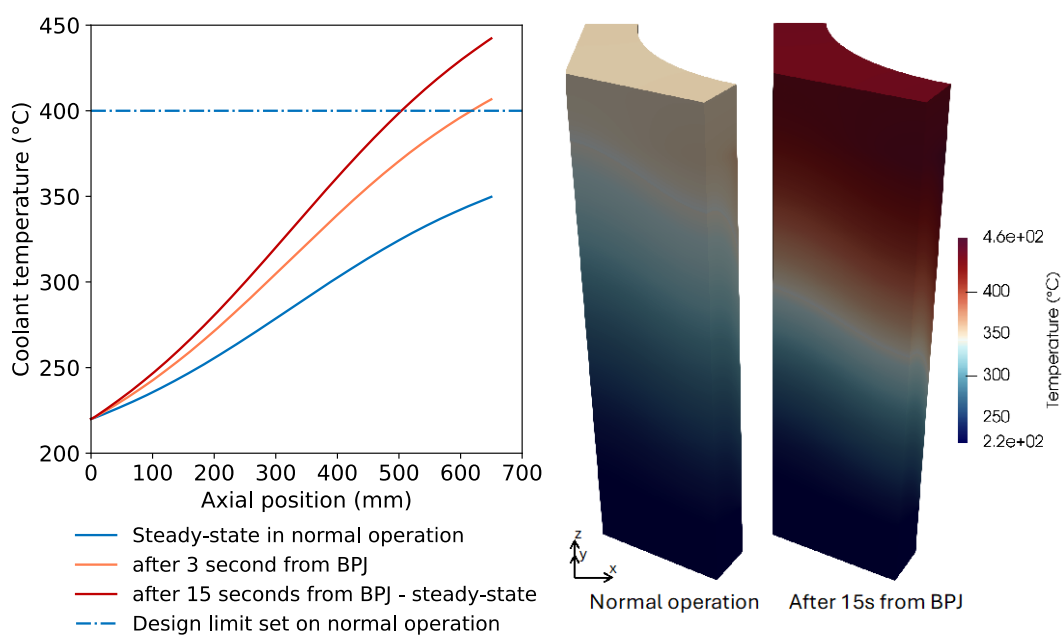


Figure 6: On the left side, the axial temperature profile of the coolant along the active length of the MYRRHA-IPS interior sub-channel is depicted at three different instants of time: before the occurrence of the BPJ (blue line), three seconds after the transient event (orange line) and fifteen seconds following the BPJ (red line). The blue dashed-dot line denotes the prescribed safety limit of 400°C during normal operation. On the right side, a contour plot illustrates the temperature distribution along the length of the sub-channel.

By employing the CFD simulation, it is also feasible to evaluate the coolant velocity within the IPS sub-channel (Figure 7), crucial for preventing the erosion of the cladding outer surface by the heavy LBE flowing. The maximum permitted value, 2 m s^{-1} , is reached far from the cladding surface, thus avoiding any erosion issues.

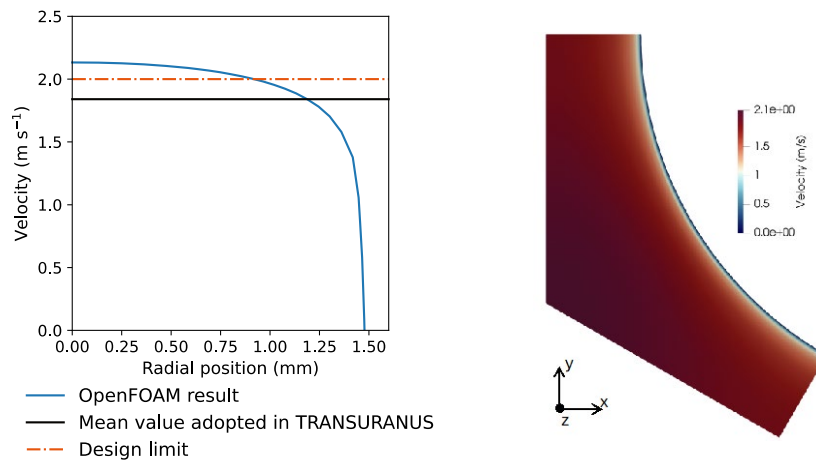


Figure 7: On the left, the radial coolant velocity profile evaluated at the midplane of the MYRRHA-IPS interior sub-channel is reported, featuring the mean value adopted in TRANSURANUS (1.84 m s^{-1}) and the design limit (2 m/s). On the right, a contour plot illustrates the velocity field at the midplane of the sub-channel. The white arrow in the contour plot highlights the velocity values depicted in the graph on the left.

The total pressure drops (gravitational and frictional) evaluated via OpenFOAM along the sub-channel length correspond to 0.114 MPa during the IPS normal operation (aligned with the semi-empirical calculation adopting the Blasius correlation for the friction factor). The coolant temperature rise determined by the BPJ transient leads to a marginal reduction in LBE viscosity and, consequently, contributes to a slight decrease in pressure drops (0.112 MPa).

Figure 8 depicts the comparison between the predicted axial cladding outer temperature profiles at different times by two simulation methods. On the left side, represented by solid lines, are the results from the fluid dynamics simulation, while on the right side, indicated by the dashed lines, are the outcomes from the TRANSURANUS simulation. In TRANSURANUS, the thermo-hydraulic calculation in the coolant relies on empirical correlations, such as the Ushakov correlation used in this work, to address the heat exchange between the coolant and the cladding. The observed discrepancy at the sub-channel inlet, peaking at approximately 22°C after 15 seconds from the BPJ event, is ascribed to the development of the thermal boundary layer, a phenomenon not accounted for in the TRANSURANUS code. The average discrepancy along the sub-channel length, without considering the discrepancy at the inlet, between CFD and TRANSURANUS computations amounts to 5°C .

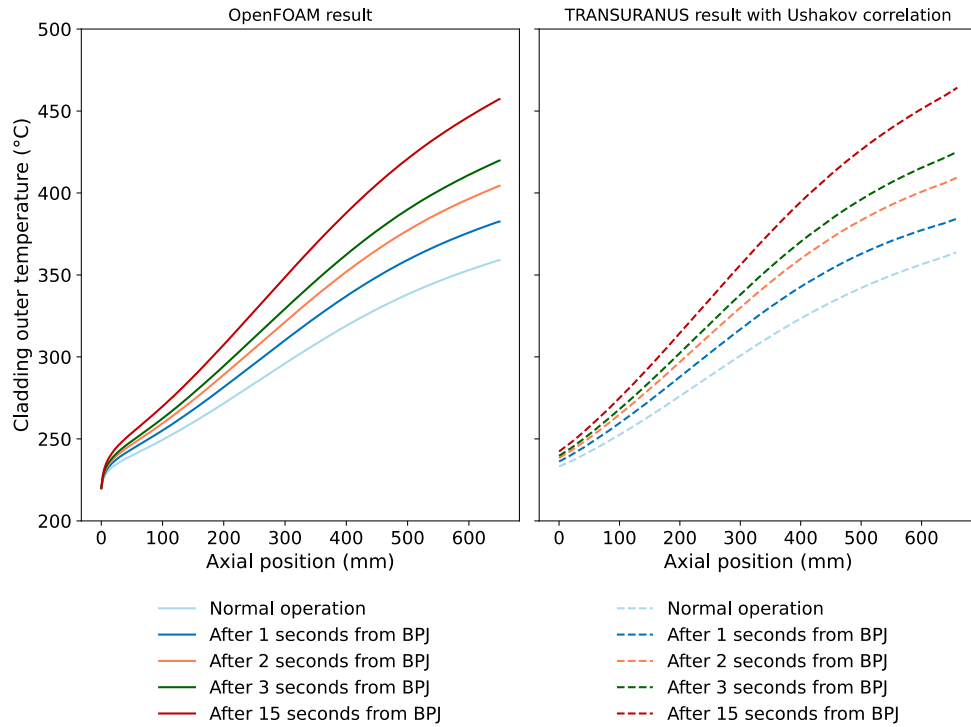


Figure 8: Evolution in time of the axial temperature profile of the cladding along the active length of the MYRRHA-IPS interior sub-channel evaluated via the OpenFOAM CFD simulation (left – solid lines) and via the TRANSURANUS code (right – dashed lines) employing the Ushakov et al. correlation, as referenced in (Ushakov et al., 1977), to model the heat transfer from the coolant to the cladding.

4.2 Fuel pin performance results

4.2.1 Base irradiation

Figure 9 showcases the evolution over time of the fuel central temperature and the gap dynamics (accounting for both fuel and cladding deformations) during the normal operation of the IPS, considering two different americium content. At the beginning of the IPS cycle, the fuel temperature is higher with respect to the end of the cycle being the gap between fuel and cladding much wider. As irradiation progresses, the fuel temperature gradually decreases while the gap narrows, owing to the greater thermal expansion of the fuel relative to the cladding. Midway through normal operation, approximately 1000 hours from the beginning of cycle, fuel densification becomes influential, resulting in a widened gap and consequently elevating the fuel central temperature. Subsequently, beyond 2000 hours, fuel gaseous swelling (here simulated via the SCIANTIX code) causes a slight reduction in the gap width, since the fission gas production itself is very low during a single cycle irradiation scenario. Upon the reactor shutdown, the fuel central temperature is reduced to 220°C (as well as the cladding and coolant temperature), determining the recovery of the fuel and cladding deformation, thereby leading to the reopening of the gap between them.

An increased americium content results in a decrease in fuel thermal conductivity, consequently leading to higher fuel central temperatures and a narrower gap width. Despite the thermal conductivity degradation caused by americium, the fuel central temperature predicted using KIT data is lower than the value predicted using SCK CEN data for the 0.49 wt.% Am scenario.

Furthermore, in any situation, the fuel central temperature never reaches the maximum allowable value of 2600°C² to avoid the fuel melting, with a safety margin of 1400°C. Additionally, no fuel-cladding mechanical interaction (gap closure) is observed, with the minimum gap reaching to 65 µm.

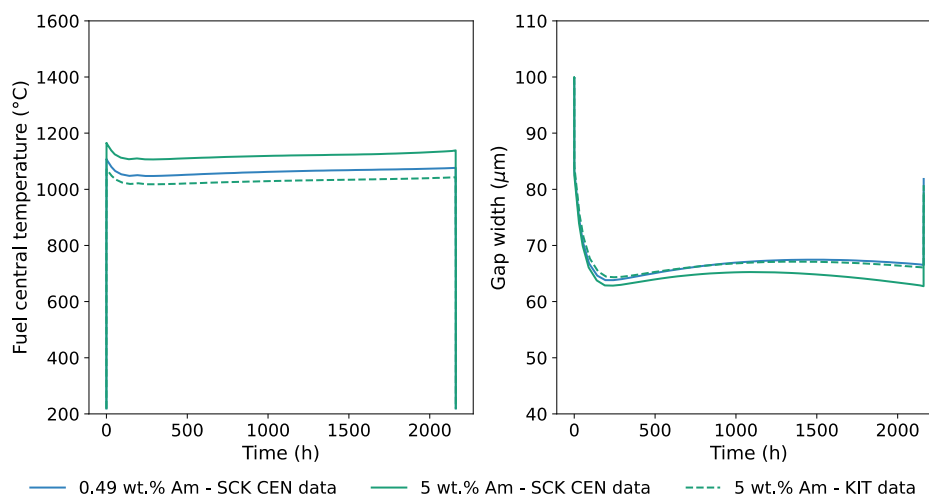


Figure 9: Evolution of the fuel central temperature (on the left) and the gap width (on the right) at the peak power node during the MYRRHA IPS irradiation conditions, as provided by SCK CEN (full line) and KIT (dashed line), for two different fuel Am contents (0.49 wt.% and 5 wt.%).

Another quantity of interest for assessing the MYRRHA fuel pin is the equivalent von Mises stress, as depicted in Figure 10. It is evidence that the americium content has no discernible impact on the stress within the cladding. The obtained stress values remain significantly below the yield stress threshold, estimated at around 400 MPa (Van Uffelen et al., 2024-1), indicating the absence of plastic deformation within the cladding during normal operation. The maximum value of the von Mises stress is located on the clad inner wall and does not exceed 60 MPa.

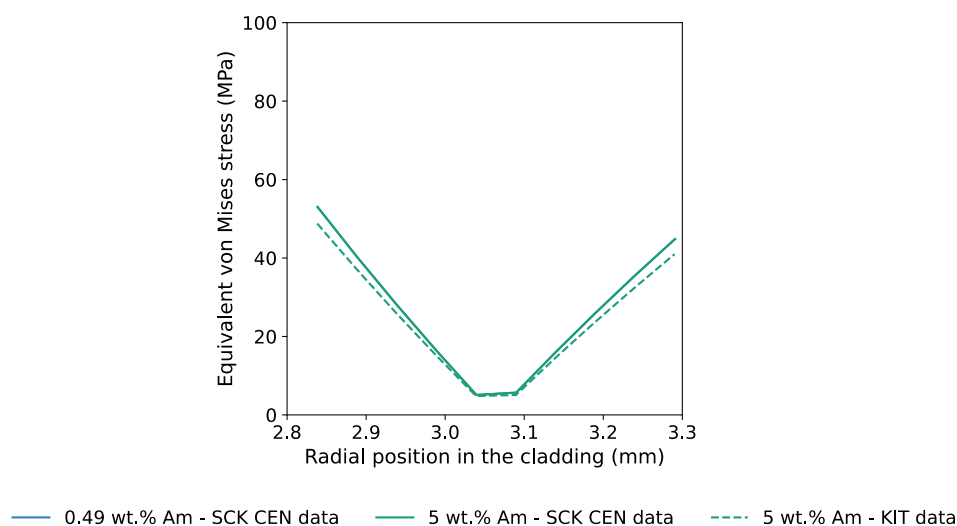


Figure 10: Radial profile of the equivalent von Mises stress in the cladding assessed at the axial location coinciding with the point of maximum temperature gradient (just beneath the ppn) and captured at the beginning of the IPS cycle, immediately following reactor start-up, as provided by SCK CEN (full line) and KIT (dashed line), for two different fuel Am contents (0.49 wt.% and 5 wt.%).

² The limit on the fuel central temperature would correspond to 2756°C, calculated using the new correlation developed in (Van Uffelen et al., 2024-2). The value adopted here (2600°C) represents a conservative value.

4.2.2 BPJ over-power transient

In accordance with Section 2, the BPJ scenario is assumed to occur at the beginning (BOC) and at the end (EOC) of the IPS cycle. The primary focus of this analysis is to examine pin performance and safety during the BPJ transient. The analyzed period corresponds only to the transient (the 1 millisecond of rapid power increase, the 3 seconds of holding power and then the 3 seconds of reactor shut-off). In Figure 11 is reported the temporal evolution of the fuel central temperature. The worst-case scenario arises when considering the maximum americium concentration (5 wt.%), attributed to the degradation of thermal conductivity as previously discussed, and considering the BPJ event occurring at the beginning of the cycle, where the fuel-cladding gap is wider compared to the end of the IPS cycle. The maximum temperature is reached 0.57 second after the initiation of the shutdown (namely, 3.57 seconds after the onset of the transient). Thus, the maximum temperature is reached after the insertion of the energy, showing the importance of assessing as well as possible the inertial effects in the fuel (storage then release of the energy). Notably, the prediction based on the data supplied by KIT consistently yields lower fuel central temperatures. Furthermore, even during the over-power transient, the safety limit on the maximum fuel temperature is maintained with a substantial margin relative to the melting point (2600°C).

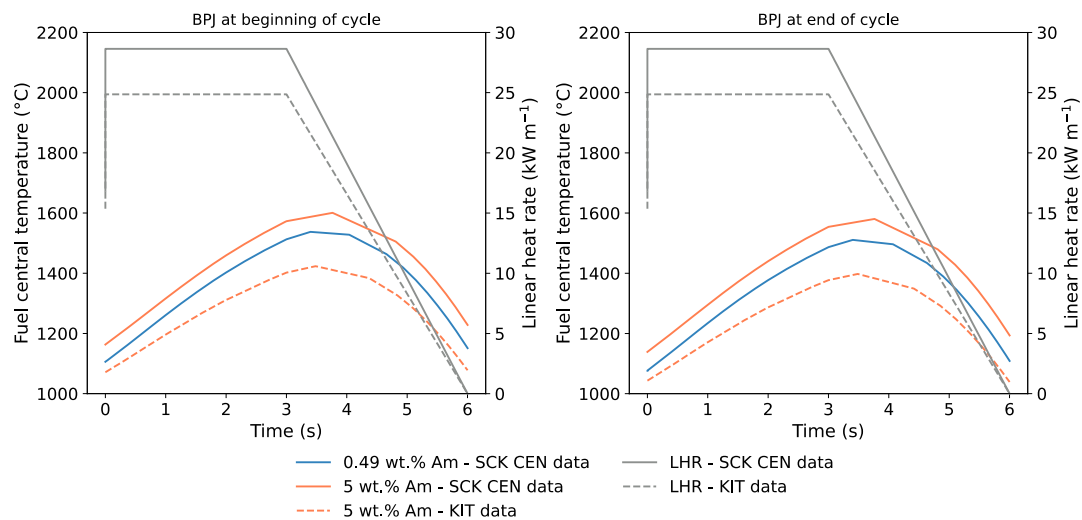


Figure 11: Evolution in time during the BPJ transient of the fuel central temperature at the peak power node considering the BPJ occurring at the beginning (left) and at the end (right) of the IPS cycle. The linear heat rate provided by SCK CEN and KIT is shown in grey.

To demonstrate the impact of the updated modelling of Am-MOX specific heat and thermal conductivity, conducted within the scope of PATRICIA Task 5.1 (Section 3.1), Figure 12 presents a comparison of the fuel central temperature with state-of-the-art models (Chawla et al., 1981; Magni et al., 2021)³. Discrepancies arise notably with a 5 wt.% Am content, primarily due to the more pronounced degradation of thermal conductivity, as observed during base irradiation in (Van Uffelen et al., 2024-1). The predicted peak fuel temperature is 1627°C, well below the design limit of 2600°C for fuel melting temperature, as established with the state-of-the-art models.

³ The model of Chawla (Chawla et al., 1981) for the specific heat holds for MOX fuel.

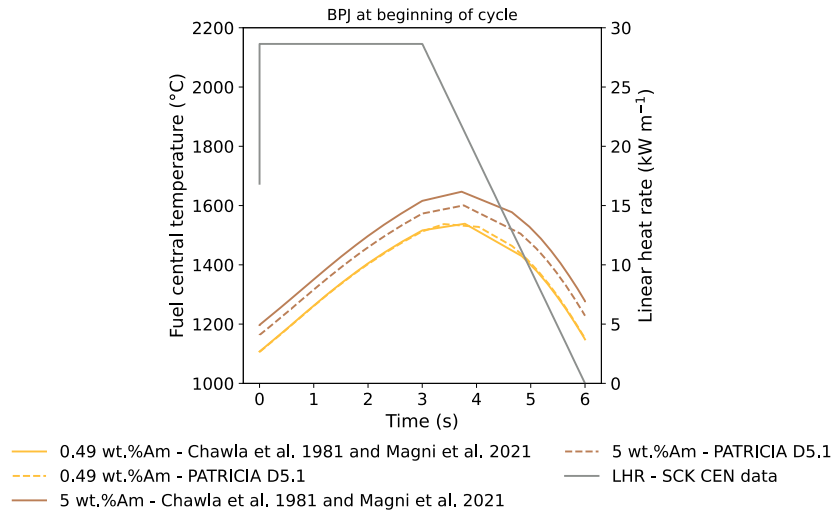


Figure 12: Evolution in time during the BPJ transient of fuel central temperature at the peak power node, as predicted by the TRANSURANUS code adopting the model of (Chawla et al., 1981) for the specific heat and the model of (Magni et al., 2021) for the thermal conductivity, against the predictions made with the updated models developed within Task 5.1 of PATRICIA Project (Van Uffelen et al., 2024-2), for two different fuel Am contents (0.49 wt.% and 5 wt.%).

Figure 13 illustrates the gap dynamics, showing that the gap consistently remains open. Unlike the behaviour of fuel central temperature (depicted in Figure 11), the most significant narrowing of the gap takes place during the BPJ transient occurring at the end of the IPS cycle. This phenomenon is ascribed to the increased gaseous swelling of the fuel. The influence of americium content on the gap width is indirect, mediated through fuel temperature: higher fuel temperatures result in greater thermal expansion of the fuel, leading to a reduction in gap width. At the onset of the transient, thermal expansion governs the dynamics of the gap, causing it to narrow until it reaches a minimum value (achieved after 3.57 seconds from the start of the transient), followed by the reopening of the gap. The time with the minimum gap size coincides with that with the maximum temperature in the fuel – and consequently with the maximum thermal expansion of the fuel.

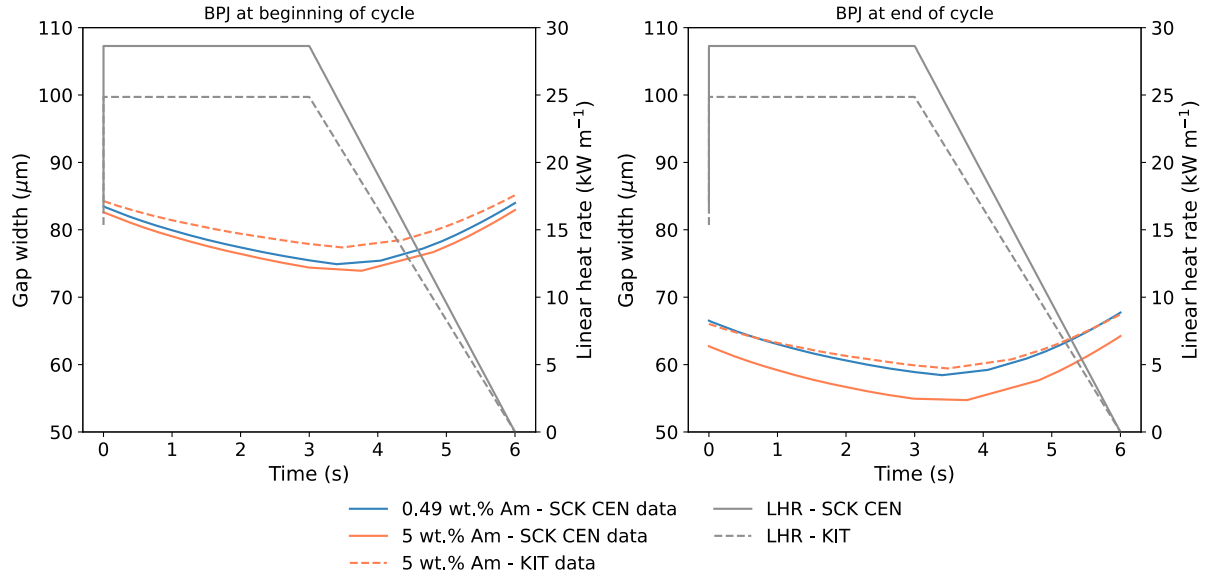


Figure 13: Evolution in time during the BPJ transient of the gap width at the peak power node considering the BPJ occurring at the BOC (left) and EOC (right). The linear heat rate provided by SCK CEN and KIT is shown in grey.

As no fuel-cladding mechanical interaction is predicted, the contributors to the cladding stress state include pressure loadings from the gas in the gap on the inner surface and from the coolant on the outer side, along with the thermal gradient across the cladding causing thermal stresses. The maximum von Mises equivalent stress predicted in the cladding during the transient is approximately 76 MPa when the BPJ occurs at BOC (Figure 14). This value is well below the yielding stress, thereby avoiding cladding plasticity by a significant margin. As previously observed during normal operation, the presence of americium content exhibits no discernible impact on the cladding stress.

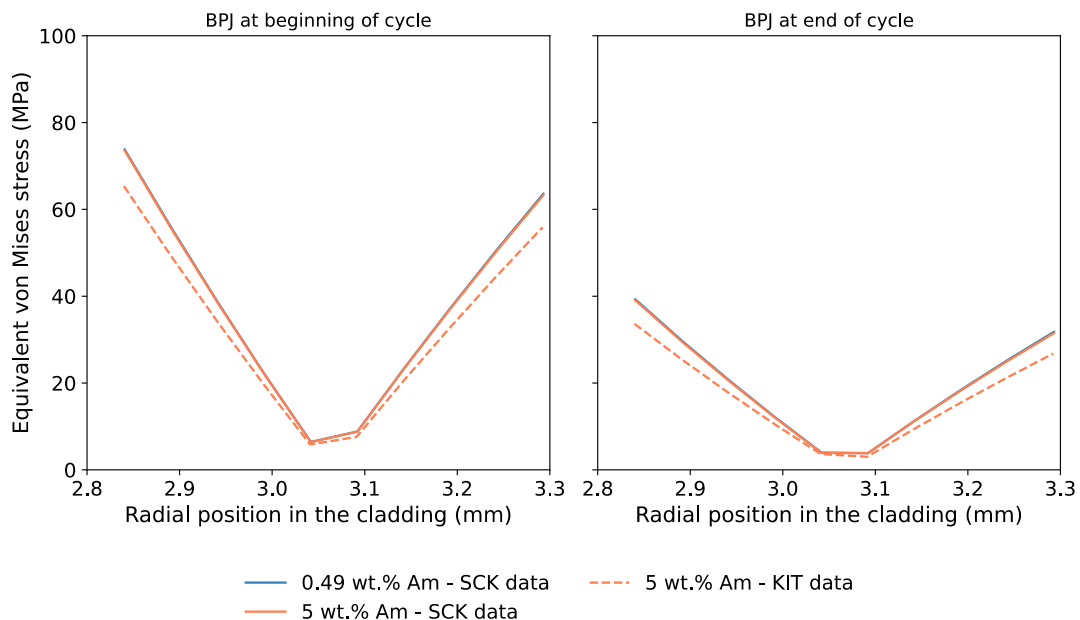


Figure 14: Radial profile of the equivalent von Mises stress in the cladding assessed at the axial location coinciding with the point of maximum temperature gradient (just beneath the ppn) and taken after 3 seconds the BPJ has occurred, for both the BPJ occurrences at the BOC (left) and EOC (right).

Figures 15 and 16 illustrate the temporal evolution of the gap pressure and the cumulative fission gas release (FGR), respectively, utilizing linear heat rate and fast neutron flux data provided by SCK CEN (see Figures 2 and 4). The behaviour of inert gases within the fuel is crucial for pin performance, as it influences the thermal conductance of the gap. These two metrics are closely intertwined: higher fission gas release leads to increased pressure within the gap. In the scenario of a BPJ at BOC, there is virtually no gas release, for the main reason that there is no gas production yet. Conversely, in the case of BPJ at EOC, an elevated americium content results in greater FGR, as the higher fuel temperature affects the gas diffusion coefficient.

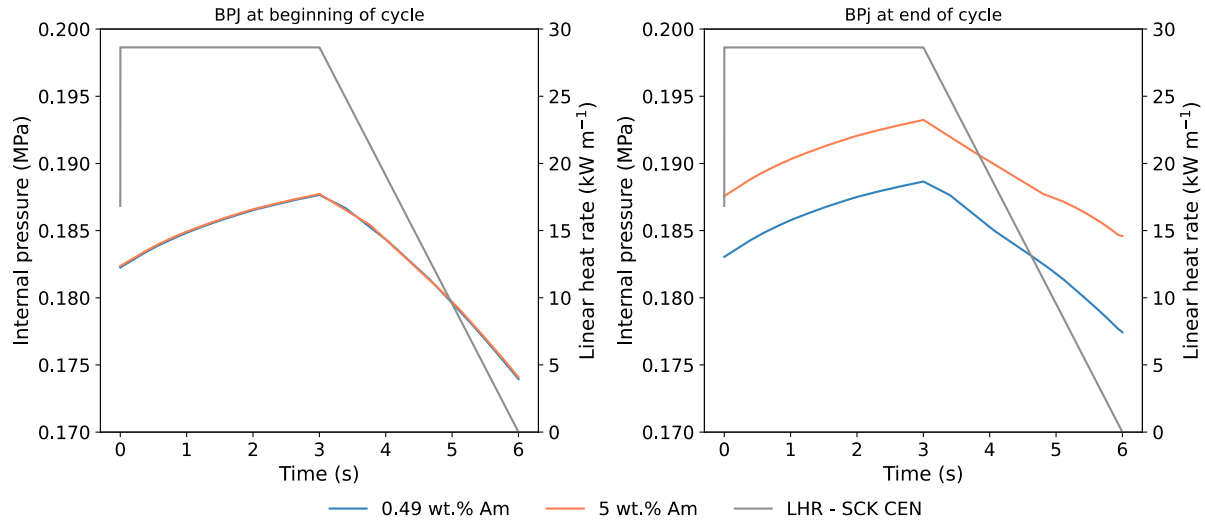


Figure 15: Evolution in time during the BPJ transient of the pin internal pressure at the peak power node considering the BPJ occurring at the BOC (left) and EOC (right). The linear heat rate provided by SCK CEN is shown in grey.

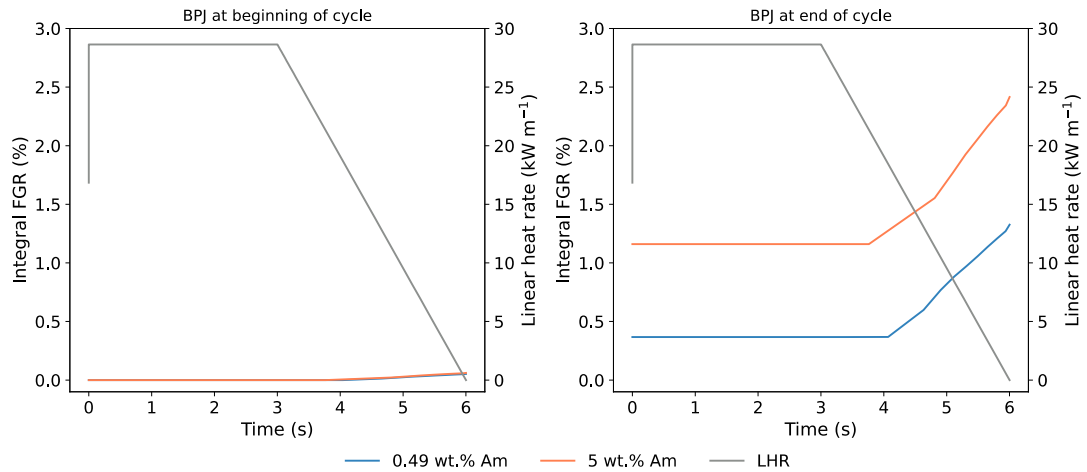


Figure 16: Evolution in time during the BPJ transient of the integral fission gas release considering the BPJ occurring at the BOC (left) and EOC (right). The linear heat rate provided by SCK CEN is shown in grey.

Figure 17 portrays a hypothetical scenario called “unprotected BPJ” (illustrated by the blue line) where the BPJ transient extends beyond 3 seconds at which should occur the scram of the reactor, compared to the “protected BPJ” (illustrated by the orange curve) considered so far where the reactor shutdown takes place after the 3 seconds. Is clearly visible the fuel temperature dynamics governed by the fuel specific heat. After 3 seconds of BPJ over-power, the fuel does not have time to reach the steady state value (reached instead after 15 seconds). Also, in this hypothetical situation the fuel central temperature is well below the design limit.

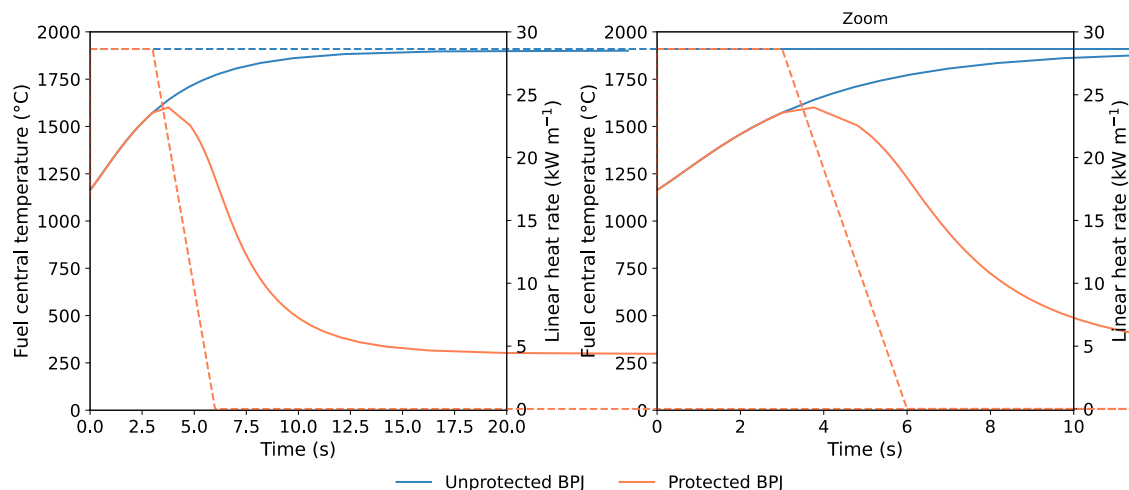


Figure 17: Evolution in time of the fuel central temperature (full line) for two different scenarios: i) maintenance of the power transient beyond 3 seconds at which should occur the reactor shutdown (blue line); ii) reactor shutdown after 3 seconds from the beginning of the BPJ (orange line). The corresponding linear heat rate is indicated by the dashed line.

5 Conclusion and future developments

In the current deliverable the performance of the hottest americium-bearing fuel pin irradiated in the in-pile test sections (IPS) sub-assembly of the MYRRHA research reactor (design revision 1.8) during the Beam Power Jump (BPJ) over-power transient (selected within the PATRICIA Project – Milestone MS5 (M. Lainet et al., 2022)), of relevance for the MYRRHA sub-critical core, is evaluated. The simulations are performed with the TRANSURANUS thermo-mechanical fuel performance code coupled with the SCIANTIX physics-based module for inert gas behaviour and incorporating the propaedeutic advancements for transient simulations pursued within the framework of PATRICIA Work Package 5.

The analysis performed confirms the safe operation of the MYRRHA IPS fuel pin in transient operative condition, being the MYRRHA safety criteria (regarding the margin to fuel melting, allowed cladding plasticity and prevented outer corrosion of the cladding) respected by wide margins. The minimum margin to fuel melting is reached, according to TRANSURANUS calculations, after 3.57 seconds from the beginning of the BPJ transient and is approximately 1000°C concerning the design limit preventing fuel melting issues. The stress in the cladding is found to be at least five times below the yield strength thus avoiding any plastic deformation of the cladding. This outcome relies on the assessment that pellet-cladding mechanical interaction can be avoided, and this corresponds to design and fabrication requirements consistent with operational strategies for the fuel pin in MYRRHA. For what concerns the

cladding thermal behaviour, any risk of outer cladding corrosion by the LBE coolant is avoided since the transient duration is limited by design to just 3 seconds.

As a complement of the beam power jump transient to be addressed in the future, the developed simulation tools and methods can be applied to beam power trips and other transient scenario connected to the accelerator driven system.

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