INSPYRE

Investigations Supporting MOX Fuel Licensing in ESNII Prototype Reactors

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D7.6 – Synthesis of INSPYRE results

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SUMMARY

This document presents a synthesis of the results of the INSPYRE H2020 European Project, which was dedicated to the investigation and simulation of uranium-plutonium mixed oxide (MOX) fuels for fast reactors and their behaviour under irradiation.

It first describes the progress in the understanding and description of the behaviour of MOX fuels obtained thanks to the combination of basic research (multiscale modelling and separate effect experiments) and examination of fuels irradiated in reactors in past experiments. Four operational issues were particularly studied: margin to fuel melting; irradiated fuel thermochemistry and interaction with the cladding; self-diffusion properties and inert gas behaviour; evolution of mechanical properties under irradiation.

Second, the advances made in the simulation of fast reactor MOX fuels thanks to the development of more physically justified models and their implementations in three European fuel performance codes are presented.

Then, the results of the simulation of two ESNII reactor cores are shown: normal operation conditions in the ASTRID sodium fast reactor prototype and normal and transient conditions in the lead-bismuth cooled reactor of the MYRRHA accelerator driven system.

Finally, the activities conducted concerning education and training, dissemination, exploitation and communication are summarized.

CONTENT

GLOSSARY

ADS	Accelerator Driven System
BPJ	Beam Power Jump
E&T	Education and Training
EPFD	Equivalent Full Power Days
ESNII	European Sustainable Nuclear Industrial Initiative
FCCI	Fuel Cladding Chemical Interaction
FCMI	Fuel Cladding Mechanical Interaction
HPC	High Performance Computing
INSPYRE	Investigations Supporting MOX Fuel Licensing in ESNII Prototype Reactors
JOG	Joint Oxyde-Gaine
JPNM	Joint Programme on Nuclear Materials
LWR	Light Water Reactor
MOX	Uranium-plutonium mixed oxide fuel
O/M	Oxygen-over-metal ratio
ROG	Réaction Oxyde-Gaine
TEM	Transmission Electron Micoscopy

1 INTRODUCTION

Fuel is at the heart of all nuclear reactor systems. Mastering the understanding of its behaviour is challenging due to the complex coupled phenomena (physical, chemical, radiation, thermal and mechanical) induced by fission. All occur in steep temperature gradients and have consequences at a multitude of length and time scales. Fuel performance predictions for licensing under normal operation and accidental conditions have relied traditionally upon extensive integral irradiation testing to generate empirical laws. Though successfully deployed for the four fast reactors operated in Europe thus far, they are not easily extrapolated to other conditions prevalent for the licensing of first uranium-plutonium mixed oxides (MOX) cores for the ESNII reactor systems.

Leveraging the knowledge from past integral irradiation testing programmes is essential to overcome the challenges of timely cost-effective licensing of ESNII first cores. The solution lies in a basic science approach to develop the intricate models underpinning the empirically derived performance laws, so that they can be extended into other operational regimes. To contribute to this objective, INSPYRE:

- Used a basic research approach combining out of pile separate effect and physical modelling and simulation to get further insight into the underlying phenomena governing the behaviour of $(U,Pu)O_2$ mixed oxide (MOX) fuels under irradiation;
- Performed additional post-irradiation examinations on selected samples to complete the results available in the literature;
- Extended the reliability regime of empirical laws describing nuclear fuel under irradiation using the results obtained in the project as well as those of previous integral neutron irradiation tests;
- Used these results to improve the reliability of European operational fuel performance codes (FPC) in normal and off-normal situations to facilitate nuclear fuel licensing and improve safety.

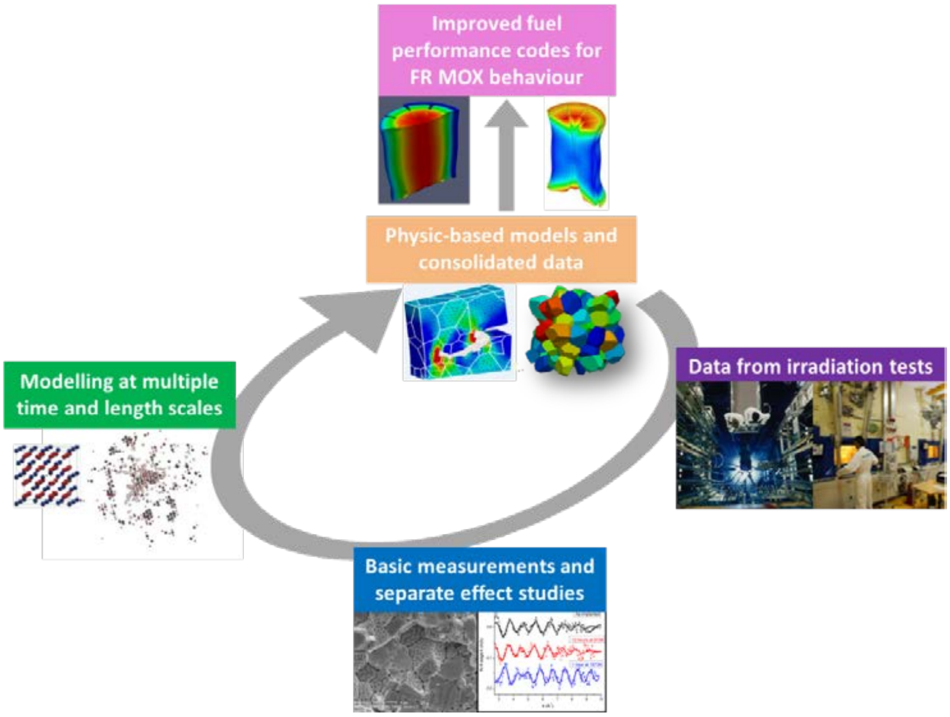

Figure 1: Approach applied in INSPYRE for the improvement of FPC for the simulation of MOX fuel.

Four important operational issues were the subject of all the investigations and developments performed in INSPYRE, from basic research to implementation in codes:

- Margin to fuel melting
- Irradiated fuel thermochemistry and interaction with the cladding
- Self-diffusion properties and inert gas behaviour
- Evolution of mechanical properties under irradiation.

In addition, INSPYRE, aimed at contributing to train the next generation of researchers on nuclear fuels and giving reliable information on nuclear fuels to the public.

2 ADVANCES IN THE UNDERSTANDING AND DESCRIPTION OF THE BEHAVIOUR OF MOX FUELS

2.1 Margin to fuel melting

One strong objective of the safety studies is to determine the margin to fuel melting since design rules indicate absence of melting as a criterion to be respected at all time, especially at the beginning of irradiation, when the power is the highest. It is all the more important for ESNII reactors, which will operate at high temperature. The knowledge of the margin to melting necessitates inter alia a comprehensive knowledge of the system phase diagrams, in particular the solid/liquid transition temperatures of the MOX as a function of the fuel parameters (composition, chemistry and burnup), but also a thorough understanding of the mechanisms governing the thermal conductivity and specific heat evolution. The effect of the americium produced by plutonium decay on these transition temperatures is also of interest.

Concerning the phase diagrams of fresh MOX, structural, energetic and melting properties of $(U,Pu)O_2$, $(Pu,Am)O_2$ and $(U,Pu,Am)O_2$ were measured using laser heating [1,2] and calculated at the atomic scale [3,4,5]. The impact of Am on these properties was also studied [6,7]. The data obtained were combined to data of the literature and of the FP7 ESNII+ project [8] to improve the thermodynamic modelling of the (U-Pu-Am-O) system [1,2,9]. Figure 2 shows the updated $UO_{1.98}$ -$PuO_{1.96}$ phase diagram compared to the experimental data available and the previous model.

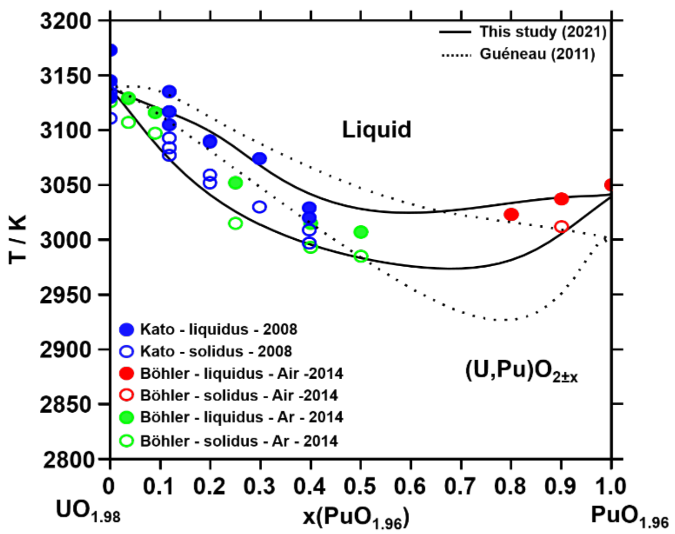

Figure 2: Updated $UO_{1.98}$ -$PuO_{1.96}$ phase diagram, experimental data available and previous model.

In addition, the existing experimental results and correlations of thermal conductivity of irradiated fast reactor MOX were reviewed and compared to LWR fuel data [10]. On this basis, parameters and mechanisms affecting the thermal conductivity of fuels in fast reactors were determined. Then, the thermal diffusivity of MOX with 30% Pu was measured up to 1400K, which yielded higher values than those of previous recommendation for MOX and closer to those for UO_2.

Finally, the specific heat of $(U,Pu)O_2$ as a function of composition (plutonium content, O/M ratio) was also evaluated using atomic scale calculations [11,12], yielding information at very high temperatures and over the full composition range, which are very difficult to measure experimentally. The results show a peak below the melting temperature caused by the Bredig transition, which is well known in compounds with fluorite structures, including UO_2, and was confirmed in PuO_2 and MOX by the atomic scale calculations.

2.2 Irradiated fuel thermochemistry and interaction with the cladding

The increasing inventory of fission products in the fuel with burnup and time has a substantial effect on the fuel chemistry, including the oxygen potential of the fuel, which governs many fuel properties. Among these fission products, volatile elements such as caesium, tellurium, iodine, as well as molybdenum, migrate from the centre to the periphery of the fuel pellet to form the "joint oxyde-gaine" (JOG) layer, observed in irradiated fuel pins beyond 7 at.% burnup. This layer constitutes a potential interaction risk between the fuel and the cladding, the fuel-cladding chemical interaction (FCCI), and results eventually in the corrosion of the cladding, the "reaction oxyde-gaine" (ROG). This corrosion is a major factor limiting the integrity, and therefore the lifetime, of the fuel pin in reactor.

INSPYRE improved significantly the knowledge on the thermochemistry of irradiated MOX, and especially on the JOG layer. First, the incorporation and transport properties of Iodine and Caesium in MOX fuel were evaluated [13]. Second, structural and thermodynamic data available on fission product compounds in the chemical system (Cs-I-Te-Mo)-(U-Pu-O) were reviewed and properties of Cs_2Te, Cs_5Te_3, Cs_2TeO_4, Cs_2MoO_4, $Cs_2Mo_2O_7$ and $Cs_2Mo_3O_{10}$ were calculated at the atomic scale or measured [14,15,16]. The thermodynamic models on the Te-U, O-Te-U, Mo-O-Te and Cs-Mo-O sub-systems were then improved using these results and first thermodynamic models of the Cs-Te-O, Cs-Te-Mo-O, U-Te and U-Te-O systems were developed [17,18].

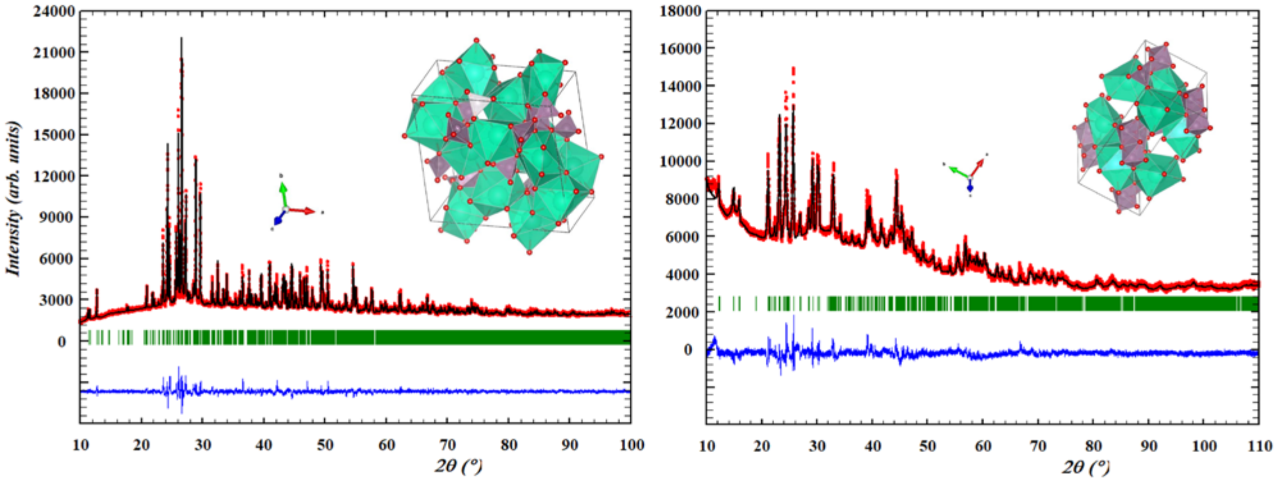

Figure 3: XRD patterns, Rietveld refinement of the data and crystal structures of (a) α-$Cs_2Mo_2O_7$ and (b) $Cs_2Mo_3O_{10}$. MoO_x polyhedra are shown in purple, CsO_x polyhedra in green and oxygen atoms in red.

Finally, in the case of a severe accident, the overheated MOX fuel and the steel of the pin cladding may interact and form a liquid phase at temperatures below the melting temperature of the fuel alone, leading to a loss of the fuel geometry and of fission product containment.

To get further insight into this key safety issue, high-temperature properties of phases formed by U, Pu, Fe and O were determined [19]. Melting temperatures in the PuO_2-Fe_3O_4 system were measured for the first time using an advanced laser heating set-up at JRC Karlsruhe. In addition, a novel experimental setup was developed at CEA Saclay to obtain new data in the liquid miscibility gap of the Fe-U-O system, in which two immiscible liquid oxides and metallic phases are coexisting. Thermodynamic models of the U-Fe-O and Pu-Fe-O systems were then derived using these experimental data.

2.3 Self-diffusion properties and inert gas behaviour

The atom transport properties of MOX fuels are at the origin of several important phenomena taking place during irradiation. First, the redistribution of Pu in the fuel strongly affects the local fuel behaviour and the oxygen diffusion governs the local oxide/metal (O/M) ratio, which has a strong impact on the thermal-chemical properties of fuels. Then, the diffusion of inert gases (fission gases, in particular xenon and krypton, and helium) governs the quantities of fission gases and helium released to the pin plenum and the pressure generated, as well as the quantity and location of the gas remaining in the fuel. Such trapped gas inside the fuel will have significant consequences on the fuel swelling and conductivity and is therefore an important factor for the thermal and mechanical behaviour of the fuel pin.

First, concerning the self-diffusion properties, atomic scale calculations, X-ray diffraction and X-Ray Absorption spectroscopy were combined to get further insight into the defects formed in $(U,Pu)O_2$, their structural, electronic and energetic properties and the influence of Am [20,21]. Then, a positron annihilation lifetime spectrometer, which will enable the detailed study of defects in Pu and Am bearing compounds, was developed in the JRC hot lab. The atomic scale study of the elementary processes governing the migration of defects in $(U,Pu)O_2$ was also initiated [22] and a new diffusion model for plutonium in MOX fuel and an associated mobility database was developed [23,24]. In addition, diffusion couples, which will enable the fine experimental characterization of diffusion coefficients of U and Pu in MOX, were designed at JRC, while an electrical conductivity was developed in the Atalante hot lab of CEA to study oxygen diffusion in oxide fuels [25].

Second, diffusion coefficients of inert gases in fresh UO_2 and MOX were determined. On the one hand, a combination of experimental techniques and modelling methods from the atomic to the grain scale were used to investigate the diffusion of Xe, Kr and He in fresh UO_2 [26]. This involved the development of a new thermo-desorption laser-heating setup enabling one to study gas diffusion and release from materials, including U-based samples, in controlled conditions at high temperature [27]. The impact of defects on gas diffusion was highlighted [28]. On the other hand, the He release from stoichiometric monocrystalline MOX samples was measured as a function of Pu content. For all samples, a release peak is observed at approximately 1900 K, i.e. a much higher temperature than for UO_2 (1100 K). These measurements enabled the determination of the diffusion coefficient of He in MOX [29].

Finally, MOX fuel samples irradiated in past campaigns containing various Pu contents and stemming from various locations in the pellet, i.e., submitted to various burnups and irradiation temperatures, were characterised using complementary techniques, inter alia transmission electron microscopy (TEM). This enabled first-of-their-kind characterizations of the microstructure of irradiated MOX at the nanometre scale. These results, combined to those of the detailed characterization of ion-irradiated and alpha-doped UO_2 [30,31,32], bring significant further insight into the mechanisms governing the evolution of the fuel microstructure under irradiation [29].

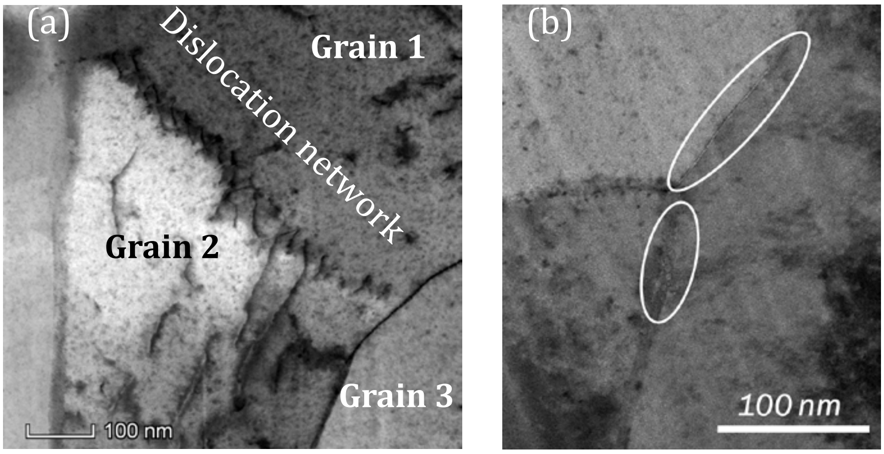

Figure 4: Bright field TEM images of a sample extracted from the high burnup structure region at the periphery of a MOX pellet irradiated in the NESTOR-3 experiment: (a) dislocation network leading to the creation of a grain boundary; (b) bubbles in grain boundaries.

2.4 Thermal-mechanical behaviour of MOX

A paramount safety issue is the integrity of the cladding, which constitutes the first retention barrier against radioactive fission product release. A design criterion is the absence or minimisation of the mechanical interaction between the fuel and the cladding. A key parameter governing the load on the cladding is the mechanical evolution of the fuel in reactor since it governs the fuel-cladding mechanical interaction (FCMI), which becomes significant during permanent low power operation or in case of a power increase after a long low-power operation. The mechanical properties also come into play in gap closure, which has a supreme influence on the temperature of the fuel and the corrosion of the cladding, but whose mechanisms are largely unknown.

To determine mechanical properties of MOX and UO_2, in particular creep and fracture properties, and get further insight into the elementary processes governing their evolution, complementary techniques and methods were combined to a precise control of materials and test conditions.

First, atomic scale calculations enabled the evaluation of the damage created in $(U,Pu)O_2$ [33], as well as the elasticity, ductile-fragile transition and ultimate strength in MOX single crystals as a function of Pu content, temperature and irradiation dose [34,35]. Mechanisms governing the evolution of mechanical fuel properties were also investigated from the atomic to the mesoscopic scale [36].

Second, a dedicated set-up was developed to determine experimentally the thermal creep in UO_2 in controlled conditions, in particular the oxygen partial pressure [37] and new micro and nano-indenters installed in the Atalante hot lab of CEA were used to measure the room temperature elasticity and micro-hardness at the grain scale of well-characterized fresh MOX fuel samples [38].

Finally, experiments were performed to investigate the irradiation-induced swelling and creep behaviour of UO_2 and MOX fuel under irradiation. First, a device was developed to measure the swelling of UO_2, which is brittle ceramic, under ion irradiation at the CNRS cyclotron in Orléans (see Figure 5a). This swelling was then measured in situ on several samples as a function of the damage dose. As shown in Figure 5b, the macroscopic swelling of UO_2 increases with the dose, in agreement with the trends shown by simulations and post irradiation examinations [39]. Second, two fuel bearing capsules were designed, fabricated and irradiated in the HFR in Petten to study the mechanical properties of fuels in situ under neutron irradiation. Deformation was recorded in MOX fuel irradiated for 6 months under very small temperature gradients. The online measurements, however, are quite scattered and will be confirmed using post-irradiation examinations after the end of the project [40].

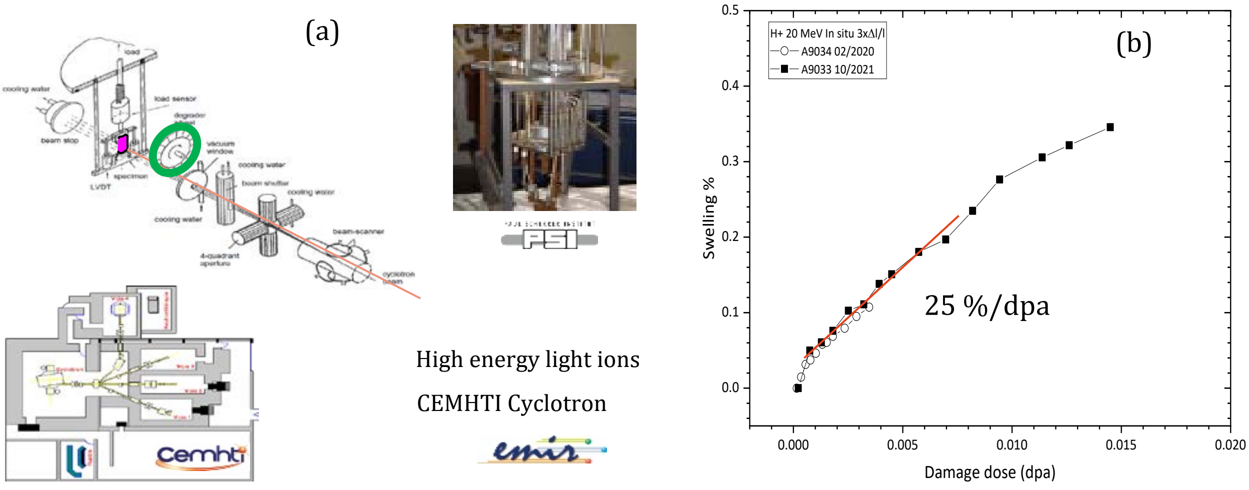

Figure 5: (a) Apparatus for on-line creep test under ion irradiation and (b) swelling as a function of damage dose on UO_2 polycristalline sample.

3 ADVANCES IN THE SIMULATION OF FAST REACTOR MOX FUELS

3.1 Development of physics-based models

Because of their largely empirical character, the models commonly used in fuel performance codes (FPC) are difficult to extrapolate to conditions different from their validation domain. One key objective of INSPYRE was to use the results obtained using basic research in INSPYRE, as well as the results from the literature, to develop more accurate and predictive models suitable for implementation in fuel performance codes. A significant part of these results obtained in the project were used to develop to new behaviour models for fuel performance codes used to simulate the behaviour of Gen IV fuels under irradiation.

First, more physically justified correlations were developed for the melting temperature and thermal conductivity of $(U,Pu)O_2$ MOX fuels, as well as for MOX including minor actinides, in conditions relevant for GEN IV reactor fuels. These include the effect of fuel temperature, Pu and minor actinide content, deviation from stoichiometry, porosity and burnup [41,42,43,44].

Second, a model for the specific heat of $(U,Pu)O_2$ based on the results of atomic scale calculations performed in the project (see section 2.1) was derived, implemented and tested in both GERMINAL and TRANSURANUS [12]. This model, shown in Figure 6, takes into account the effect of the variation in composition (Pu content and O/M ratio) at high temperature. It includes the peak below the melting temperature caused by the Bredig transition predicted by the atomic scale calculations.

Then, the modelling of fission gas and helium behaviour in MOX fuels was improved significantly thanks to the development of a significant number of physics-based models [45,46] and their implementation in the SCIANTIX grain-scale module [47]:

- a burnup model evaluating the helium production in MOX fuels [48];
- a reduced order model of the fission gas diffusion in columnar grains [49];

- a model for the intra-granular helium behaviour accounting for helium diffusivity and solubility [50,51,52] and including its interaction with fission gases [53,54,55];
- an improved description of the formation of the High Burnup Structure and of the corresponding porosity evolution at the periphery of fuel pellets [56,57].

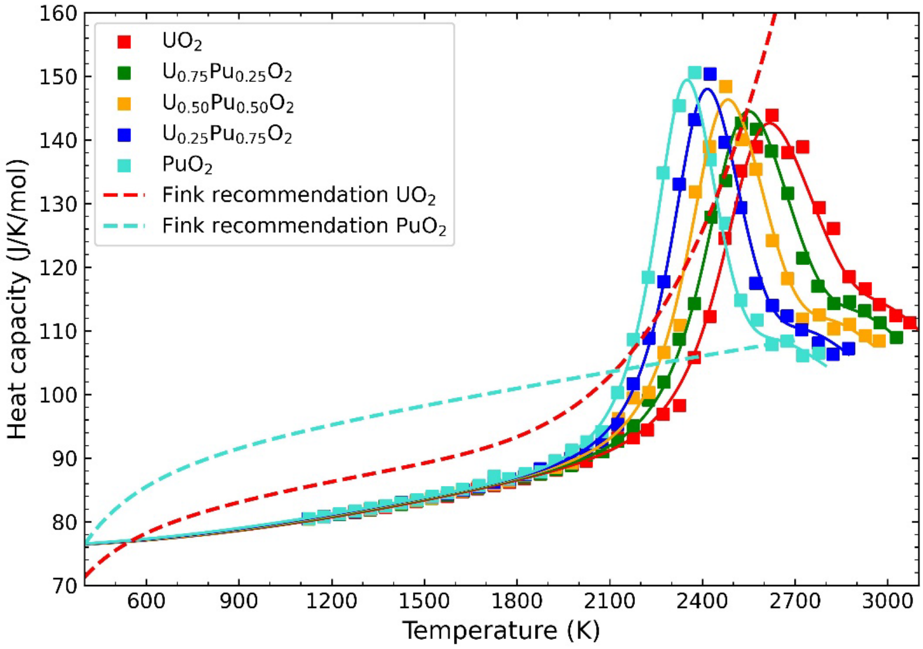

Figure 6: Heat capacity of (U,Pu)O_2 as a function of temperature and Pu content yielded by atomic scale calculations and analytical law (solid lines) fitted on these results, compared to the commonly used recommendations for UO_2 and PuO_2 by Fink [58].

Concerning mechanical properties, improved correlations for the MOX thermal expansion and Young's modulus including the impact of fuel temperature, Pu content, porosity and deviation from stoichiometry were developed using recent experimental and modelling data [59]. A methodology was also developed to describe the creep behaviour of a polycrystalline microstructure. Finally, an advanced micro-mechanical model suitable for both normal operation and off-normal conditions and including a description of the fuel rupture behaviour was also developed based on three-dimensional representative volume elements [60].

3.2 Implementation in fuel performance codes and assessment

The data obtained and the models developed in the project were used to improve three European fuel performance codes (FPC): GERMINAL, developed by CEA [61], MACROS, developed by SCK.CEN [62] and TRANSURANUS, developed by JRC with the support of academic organisations, including POLIMI [63]. The correlations for MOX thermal-mechanical properties (thermal conductivity, melting temperature, thermal expansion, Young's modulus) obtained in the project were implemented in the three codes and these were coupled with the SCIANTIX grain-scale module for inert gas behaviour [64].

The improved versions of the codes were assessed on the simulation of the three past fast-neutron irradiation experiments SUPERFACT-1, RAPSODIE-I and NESTOR-3 [65]. The code validation against local and integral experimental data [66,67] reveals harmonized predictions of fuel temperature and restructuring between the various codes, as well as better end-of-life values of fission gas release compared to the pre-INSPYRE versions [68], as shown in Figure 7. This represents a first proof of the effectiveness of the inclusion of physics-based models in FPCs.

(a)

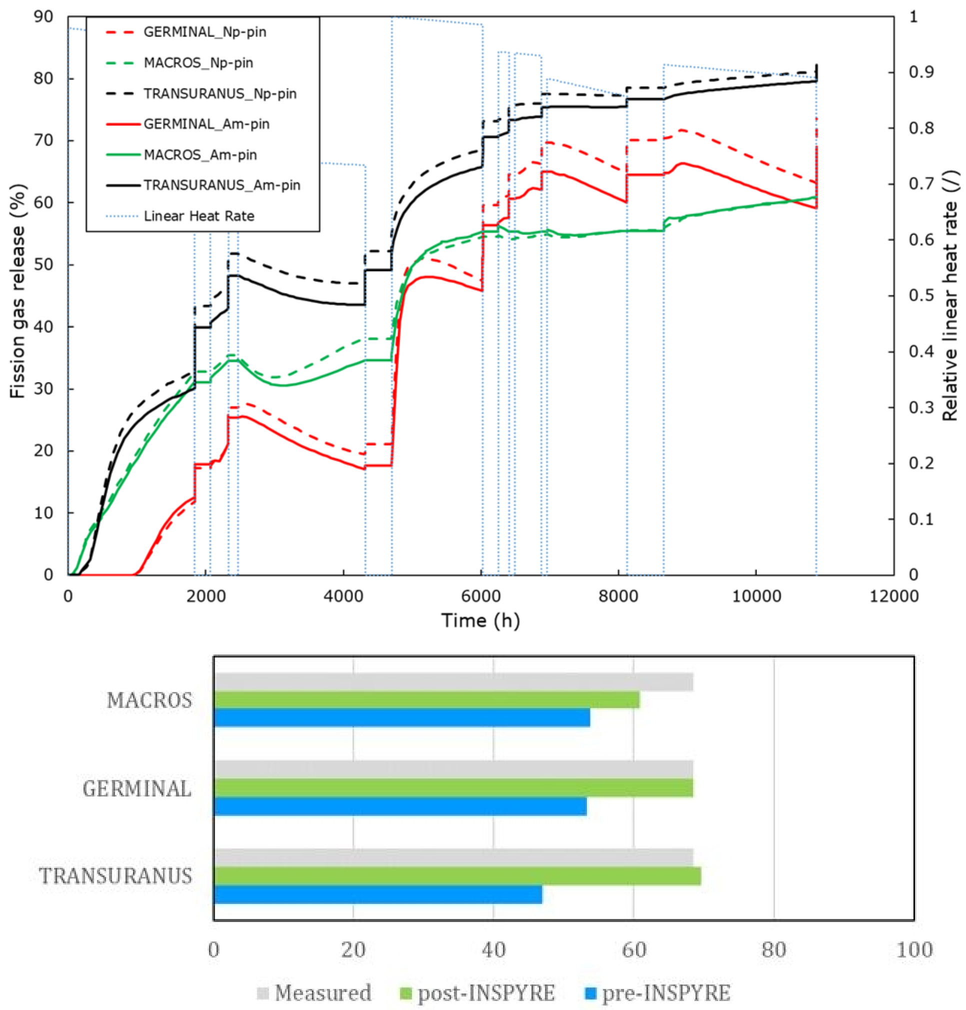

Figure 7: Results of the simulation of the SUPERFACT-1 irradiation (a) Relative linear rate and fission gas release at peak power node in SUPERFACT-1 irradiation yielded by improved FPC (b) Integral fission gas release yielded by pre and post-INSPYRE versions of FPC and comparison with experimental data.

4 SIMULATION OF ESNII REACTOR CORES

The final objective of INSPYRE was to apply the extended FPCs to the simulation of ESNII fuel elements to support their design and licensing. The post-INSPYRE versions of GERMINAL, TRANSURANUS and MACROS were used to simulate normal operation conditions in the ASTRID sodium fast reactor prototype [69], as well as normal and transient conditions in the lead-bismuth cooled reactor of the MYRRHA accelerator driven system (ADS) [70]. The results were also compared to those of the pre-INSPYRE versions.

4.1 Simulation of ASTRID normal operating conditions

The ASTRID fuel pin concept is based on annular pellets and a heterogeneous axial fuel column containing fertile and fissile (containing MOX) zones [71]. The nominal conditions simulated are those proposed at the end of the conceptual design phase for an operation at 1500 MW_{th} [72]. The sodium inlet temperature is assumed constant in time and equal to 400°C, with a pressure of 0.3 MPa. The irradiation history considered presents a maximum linear heat rate of 463 W/cm at the beginning of irradiation with a progressive power decrease of 10% during the whole residence time, which is of 1440 equivalent full power days (EPFD), divided in four cycles of 360 EPFD each.

This irradiation history was simulated using the pre- and post-INSPYRE versions of GERMINAL and TRANSURANUS [73,74]. These simulations are the first ones of the ASTRID fuel element available in the open literature. Key criteria to be assessed under nominal irradiation were (1) the power-to-melting margin at the beginning of irradiation and (2) the cladding maximal stress at the end of irradiation.

Figure 8 shows the fuel central temperature evolution at the peak power node yielded by GERMINAL and TRANSURANUS. It is observed that the improved thermal laws modify only slightly the results of GERMINAL, while they improve significantly the temperatures yielded by TRANSURANUS. All the code versions confirm that the temperature is well below the melting temperature during the whole irradiation.

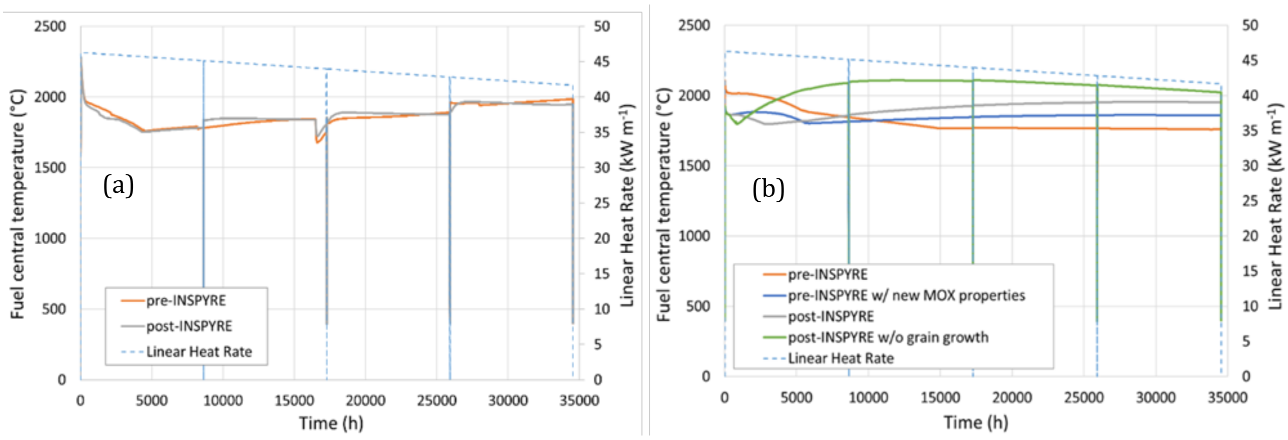

Figure 8: Fuel central temperature evolution at the peak power node yielded by (a) GERMINAL and (b) TRANSURANUS.

Concerning the pellet-cladding mechanical interaction (PCMI) and the subsequent mechanical stress in the cladding, on the contrary, significant differences are observed between the pre- and post-INSPYRE version on the one hand, and between GERMINAL and TRANSURANUS on the other hand. This difference is mainly due to a different treatment of the fuel relocation during irradiation and its accommodation between the various versions of the codes. The pre-INSPYRE version of GERMINAL yields negligible stresses induced by the pellet-cladding interaction even at the end of irradiation. This is due to the possibility of a complete accommodation of the relocation displacement in case of a closed gap and to the fact that the contribution of the gaseous swelling is neglected, assumptions which were verified on a large experimental database. In the post-INSPYRE results, the significant increase of the gaseous swelling contribution leads to a complete relocation accommodation and to a subsequent pellet-cladding interaction. TRANSURANUS does not consider any recovery of the relocation, which leads to a much earlier pellet-cladding mechanical interaction.

Finally, the coupling of TRANSURANUS and GERMINAL with SCIANTIX enabled the inclusion in the codes of a physics-based model for fission gas behaviour, which can be continuously improved when further insight on the mechanisms governing this behaviour is obtained. The results obtained highlight the need for the development of consistent models for the phenomena connected to the fission gas behaviour, e.g., for grain growth in fast reactor conditions, as well as of specific experiments to prove the improvement of the predictive capabilities of the codes concerning the pellet-cladding mechanical interaction.

4.2 Simulation of MYRRHA normal operating and transient conditions

The MYRRHA fuel pin consists in MOX fuel pellets in a cylindrical steel cladding [75] with insulator segments in yttria-stabilised zirconia ceramics. A typical MYRRHA operating schedule consists of 90 effective full power days followed by 30 days of shutdown for core reshuffling, loading and maintenance [76,77].

The pre and post-INSPYRE versions of the GERMINAL, TRANSURANUS and MACROS codes were used to simulate a nominal MYRRHA irradiation history, as well as beam power jump (BPJ) transients (protected over-power transient) occurring at the beginning and at the end of irradiation [78,79]. These BPJ scenarios are identified among the most critical for the fuel maximal temperature and cladding plasticity caused by fuel-cladding gap closure and subsequent mechanical interaction between pellet and cladding (PCMI). The aim of this study is the comparison of the simulation results with the design limits set for the MYRRHA fuel pins in terms of maximal fuel temperature and maximal allowed cladding plasticity. These simulations represent the first application of fuel performance codes to the MYRRHA pin performance at European level.

The various versions of the codes yield quite different results in terms of temperature. In particular, the temperatures yielded by the post-INSPYRE versions are significantly lower than those of the pre-INSPYRE ones. This is due to the combined effect of a reduced fission gas release, which induces a higher gap conductance, and of a higher fuel thermal conductivity, less degraded by fuel burnup. As expected, the worst BPJ scenario in term of fuel temperature is the jump occurring during the first cycle of the MYRRHA, where the reactor power is the highest. The highest temperature (2510°C) is yielded by the pre-INSPYRE version of TRANSURANUS, almost 100°C lower than the design limit conservatively set at 2600°C.

Concerning the stress level on the cladding, the highest values are obtained for a BPJ during the 12th cycle, when the fuel-cladding gap closure is the most probable. The worst scenario in term of gap closure is yielded by the pre-INSPYRE version of GERMINAL, which obtains a closed gap since the end of the 6th cycle. As seen on Figure 9, however, the stresses on the cladding calculated by all the codes, are all well below the yield stress of the cladding steel in the simulated conditions. The MYRRHA cladding therefore always remains in the elastic regime.

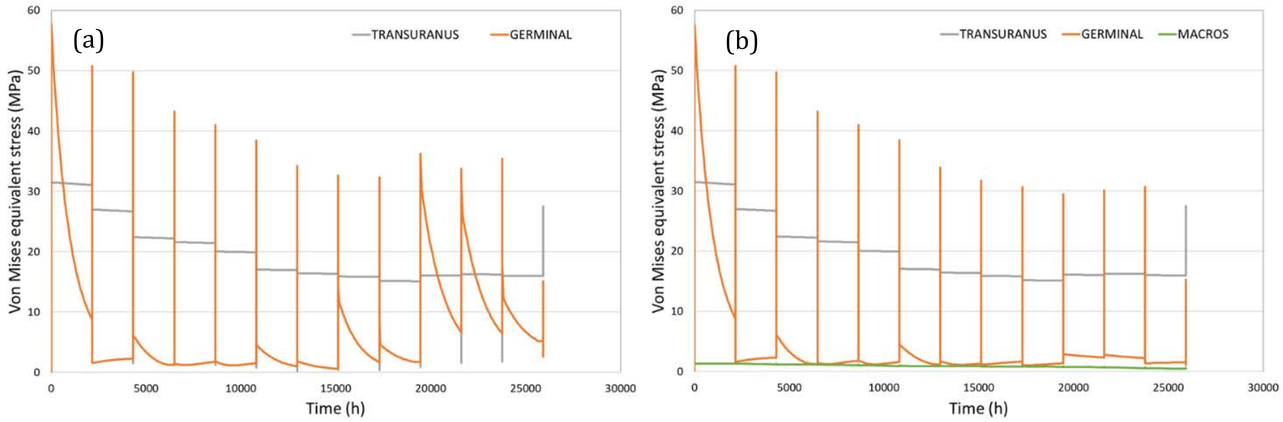

Figure 9: Comparison between Von Mises equivalent stresses in the cladding at peak power node yielded by the (a) pre-INSPYRE and (b) post-INSPYRE versions of the codes.

The various versions of the codes therefore all predict that adequate margins are respected in terms of temperature and cladding plasticity during the whole residence time, even for the hottest fuel pin and in the case of beam power jump transients.

As in the case of ASTRID fuel elements, these results constitute a firm basis for further work. Improved models, in particular for fuel relocation, grain growth and behaviour of volatile fission products, adapted to fast reactor conditions are needed to improve the predictive capabilities of the codes.

5 EDUCATION AND TRAINING, DISSEMINATION, EXPLOITATION AND COMMUNICATION

5.1 Education and training activities

INSPYRE included an ambitious training-through-research programme and 16 PhD theses on the activities of the project were defended [80]. Then, an exchange scheme was organised to foster the mobility of researchers between partner institutes and encourage collaborations, as well as to create opportunities for young scientists to acquire experience in hot laboratories and dedicated facilities. Eight exchanges took place in total, which amounted to 33 months, despite the COVID pandemic [81].

In addition, INSPYRE contributed to the effort on education of young scientists with the co-organisation of two schools together with other European initiatives. The first school on the MOX nuclear fuel cycle was organized with the GENIORS project [82] and took place from May 13 to 17, 2019 in Delft (The Netherlands) [83]. The European School on Nuclear Materials Science 2020 was organised jointly with the H2020 GEMMA [84], M4F [85] and IL TROVATORE [86] projects under the auspices of the EERA-JPNM [87] and was held online from November 8 to 13, 2020 [88].

5.2 Dissemination of the project's results

INSPYRE yielded 40 technical deliverables available to the public on the project's website [89]. More than 50 articles have been submitted or published so far on INSPYRE results and are also all available on the project's website [90]. Approximately 90 communications were presented in conferences and workshops.

In addition, yearly electronic newsletters were distributed to the nuclear materials and fuels research community to inform them of the latest achievements in the project and give them relevant news [91].

Finally, two scientific workshops were co-organised to ensure the exchange on the activities and results of INSPYRE between the participants in the project with the international nuclear fuel research community: NuFuel-MMSNF 2019 (PSI, Switzerland) [92] and NuFuel 2021 (Bangor University, United Kingdom) [93]. Then, a final international workshop was held in November 2021 in Marseille, France and online to present the results obtained in the Project to the nuclear fuel community [94]. It gathered 65 participants, including researchers from the project's partners, members of the User Group and the Scientific Advisory Committee of the project, as well as the coordinator of the EERA-JPNM and colleagues from Westinghouse, OntarioTech University, Idaho and Los Alamos National Laboratories.

5.3 Exploitation

INSPYRE created a user group composed of key customers for the project's results. It included representatives of the designers of the ESNII reactor concepts (ASTRID, MYRRHA, ALFRED, ALLEGRO), as well as of fuel manufacturers and utilities (ORANO, EDF). The approach proposed for the improvement of fuel performance codes and the knowledge gained during the project, as well as the tools developed, were presented to the member of the user group during regular meetings throughout the project [95,96]. These exchanges ensured that the innovative approach developed in the project and the results obtained are transferred to the main customers of the project. It also helps the needs of these customers to be better known to the fuel research community and considered in further studies. Finally, it contributes to convincing the users to use the codes improved during the project and to adopt progressively the approach developed, which will eventually contribute to accelerate the qualification of future fuels.

5.4 Outreach activities

Finally, outreach activities contributed to inform the public on the research conducted on nuclear energy, which is part of the answer to the challenges faced by Europe concerning energy needs and sustainability. Exhibition material for general audience consisting of five posters on the energy mix, fission reactions, nuclear reactors, nuclear fuels and simulation codes was prepared [97] and a video summarizing the objectives and results of INSPYRE [98] were released.

Training and communication materials can be reused to inform the scientific community and the general public on nuclear energy and nuclear fuels, as well as on the role simulation can play in improving the safety and sustainability of nuclear energy.

6 SUMMARY AND OUTLOOKS

Thanks to the combination of basic and applied research, INSPYRE brought significant advances on operational issues essential for the safety assessment and qualification of mixed oxide (MOX) fuels for future nuclear reactors, as well as on the simulation of the fuel behaviour in reactor.

The new experimental and modelling results obtained constitute a significant and consistent contribution to the knowledge and understanding of the physical, thermal, chemical and mechanical properties of MOX fuels. The research conducted in INSPYRE also has a scientific and technical impact: to solve the very complex issue of fuel behaviour under irradiation, INSPYRE needed excellent science and has thus driven the need for improved experimental techniques and modelling methods at all scales. A large number of set-ups enabling the detailed characterization of UO_2 and plutonium and americium bearing fuels were developed and commissioned in the hot laboratories of the partners. This permits very detailed characterization and yields results of unprecedented quality. First-of-a-kind atomic scale calculations of complex properties on complex compositions of MOX fuels were also performed.

Then, significant progress was made regarding the predictive capability of changes in material properties and behaviour in operational conditions, which has brought refinement to three European Generation-IV reactor design codes GERMINAL, TRANSURANUS and MACROS. The transfer of information and codes to manufacturers and operators guarantees crucial benefits in overcoming the bottlenecks in the certification of materials safety of GEN IV systems.

Finally, the post-INSPYRE versions of GERMINAL, TRANSURANUS and MACROS were used to simulate normal operation conditions in the ASTRID sodium fast reactor prototype and normal and transient conditions in the lead-bismuth cooled reactor of the MYRRHA accelerator driven system. The results of these simulations, which are the first openly available, enabled the evaluation of the safety margins of the fuel designs, e.g., the margin to fuel melting or the cladding plasticity for these two systems in representative operational conditions.

The work performed in INSPYRE must continue to bring further improvements in the reliability and predictive character of FPC on fast reactor MOX fuels, enabling significant advances to the licensing of MOX fuel under normal and off normal conditions. The approach developed for the improvement of fuel performance codes is expected to become a new standard. It was included in the strategic research agenda of the EERA-JPNM [99] and is now part of the strategic research and innovation agenda on nuclear materials for all reactor generations developed in the ORIENT-NM project [100].

In addition, the research performed in INSPYRE and its outcomes have already been the basis for the pilot project RODEO (Research on oxide fuels for data to improve codes) proposed in the EERA-JPNM, which should start in 2023, as well as for two European projects concerning the simulation of nuclear fuels:

- Domain 2 of the PATRICIA project (Partitioning And Transmuter Research Initiative in a Collaborative Innovation Action) [101] concerning the simulation of the in-pile behaviour of Am-bearing fuels, which started in October 2020
- OperaHPC (OPEn HPC theRmomechanical tools for the development of eAtf fuels) [102] concerning the development of open source tools for the simulation of fuel behaviour under irradiation and their application to the simulation of accident-tolerant fuel elements, which will start in November 2022.

Finally, a second edition of the European School on Nuclear Materials Science will take place in November 2022 [103]. This school is envisaged to become a regularly occurring event to advance further the training of researchers in this essential field for the development of nuclear energy.

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