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The MARINE experiment: Irradiation of sphere-pac fuel and pellets of UO_{2-x} for americium breading blanket concept

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HIGHLIGHTS

- MARINE is designed to check the behaviour of MABB sphere-pac concept.
- MABB sphere-pac are compared with MABB pellet.
- Swelling and helium release behaviour will be the main output of the experiment.
- An experiment to check sphere-pac MADF fuel behaviour has been already performed.

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ABSTRACT

Americium is a strong contributor to the long term radiotoxicity of high activity nuclear waste. Transmutation by irradiation in nuclear reactors of long-lived nuclides like ^{241}Am is therefore an option for the reduction of radiotoxicity and heat production of waste packages to be stored in a repository. The MARINE irradiation experiment is the latest of a series of European experiments on americium transmutation (e.g. EFTTRA-T4, EFTTRA-T4bis, HELIOS, MARIOS, SPHERE) performed in the High Flux Reactor (HFR). The MARINE experiment is developed and carried out in the framework of the collaborative research project PELGRIMM of the EURATOM 7th Framework Programme (FP7). During the past years of experimental works in the field of transmutation and tests of innovative nuclear fuels, the release or trapping of helium as well as swelling have been shown to be the key issues for the design of such kind of fuel both as drivers and even more for Am-bearing blanket targets (due to the higher Am contents). The main objective of the MARINE experiment is to study the in-pile behaviour of uranium oxide fuel containing 13% of americium and to compare the behaviour of sphere-pac versus pellet fuel, in particular the role of microstructure and temperature on fission gas release and He on fuel swelling.

The MARINE experiment will be irradiated in 2016 in the HFR in Petten (The Netherlands) and is expected to be completed in spring 2017.

This paper discusses the rationale and objective of the MARINE experiment and provides a general description of its design for which some innovative features have been adopted.

1. Introduction

Minor Actinides elements contribute to a large part to the long term radiotoxicity of spent nuclear fuels. Transmutation by irradiation in nuclear reactors of long-lived nuclides like ^{241}Am is, therefore, an option for the reduction of the most radiotoxic of nuclear. In the frame of the EURATOM 7th Framework Programme (FP7), the MARINE irradiation test is part of the collaborative research project PELGRIMM (Delage et al., 2012) (PELlets versus GRanulates: Irradiation, Manufacturing & Modelling), which is focussed on MA-bearing oxide fuel developments for fast reactor systems in support of the European Sustainable Nuclear Energy-Technology Platform (SNE-TP) Strategic Research Agenda in particular to the European Nuclear Industrial Initiative (ESNII). The PELGRIMM project aims to investigate two fuel forms (pellet and sphere-pac) addressing:

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- Fabrication process development as well as property measurements.
- Semi-integral irradiation assessment of MA-bearing fuels and Post Irradiation Examinations.
- Irradiation behaviour modelling and predictive code developments.

Sphere-pac technology (that leads to production of beads that can be directly loaded in pins) is attractive for MA-bearing fuels as it would lead to a significant simplification of the fabrication process and to a better accommodation of solid swelling (compared to pellets) under irradiation. Moreover, especially when dealing with highly radioactive minor actinides, dust-free fabrication processes are essential to reduce the risk of contamination of the preparation facilities.

Based on historical experience and knowledge, two MA-recycle modes are currently under consideration (Chauvin et al., 2014):

- the homogeneous recycle mode, where small quantities (<5%) of MA are mixed to the MOX driver fuel of the reactor; known as Minor Actinide bearing driver (MABD) fuels.
- the heterogeneous mode, where high MA content (10–20%) are added to an UO_2 support, with these MA-bearing sub-assemblies being located at the periphery of the core; known as Minor Actinide bearing blanket (MABB) fuels.

MARINE is the most recent of a series of irradiation tests dealing with heterogeneous recycling of minor actinides (MAs) for fast reactors (i.e. the MA-bearing blanket fuels concept, MABB). We know from previous irradiation (e.g. EFTTRA-T4 (Konings et al., 2000), EFTTRA-T4bis (Klaassen et al., 2003), HELIOS (D'Agata et al., 2011), MARIOS (D'Agata et al., 2013), SPHERE (D'Agata et al., 2014)) that the production of He due to the reaction chain of Am (see Fig. 1) has an impact on the retention and releases of gases as well as on the swelling of the fuel. The swelling and the release behaviour of sphere-pac fuel compared with pellet form, especially for fuel containing high amounts of Am has hardly been investigated at all.

MARINE consists of two pins containing ^{241}Am, based on a natural uranium oxide matrix, with two different fuel forms, pellets and sphere-pac fuel. The two fuels will have a similar heavy atoms density. Nevertheless, due to the geometry of the fuel, the sphere-pac fuel will have a lower "total" density. The resulting total density (fuel density and stacking density) is called smeared density. Both pins containing the fuels have been equipped with pressure transducers allowing the online measurement of the pressure to better understand the gas release behaviour.

The MARINE irradiation experiment is being carried out in position H6 of the HFR core, which has a thermal flux (<0.625 eV) of about 0.6×10^{18} m^{-2} s^{-1}, an epithermal flux (0.625 eV < E < 0.82 - MeV) of about 1.32×10^{18} m^{-2} s^{-1} and a fast flux (>0.82 MeV) of about 0.5×10^{18} m^{-2} s^{-1} with a total neutron flux of about 2.4×10^{18} m^{-2} s^{-1}. The neutron spectrum of the HFR does not match a typical spectrum in the blanket of a fast reactor. For this reason, the experiment is sometimes called "semi-integral" because some parameters are not representative. Nevertheless, the MARINE experiment simulate the corresponding Sodium Fast Reactor (SFR) conditions at End of Life (EoL) for an hot pin located in the SFR AmBB sub-assembly (i.e. a pin situated close to the core): @2000 EFPD, 70 W/cm (LHR), 1000 °C (fuel max temp) and 4 mg cm^{-3} of He production (Valentin et al., 2009). This paper presents the design and the objective of the experiment which was optimised to irradiate the fuel at a temperature of about 1000 °C. The experiment has been designed to last 19 cycles (considering 4 cycles of stop and 15 cycles of irradiation) but it will reach its target earlier. The post irradiation examination of the fuel irradiated in MARINE is still not planned and will be probably conducted within the framework of the Joint Programme on Nuclear Materials (JPNM) of the European Energy Research Alliance (EERA).

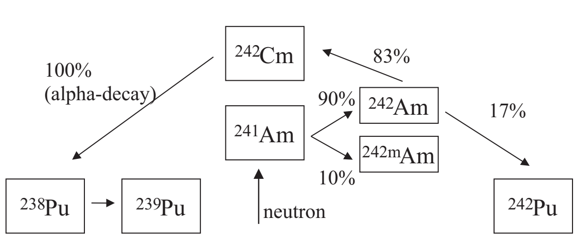

Fig. 1. Reaction chain of ^{241}Am.

2. Fuel and pin characteristics

The americium-containing fuel for MARINE, both pellet-type and sphere-pac-type, were prepared at JRC-ITU in Germany. The MARINE experiment consists of two pins of 15–15 Ti steel (see Fig. 2) with ID of 5.65 mm and OD of 6.55 mm, an austenitic steel clad which was used in the French sodium fast reactors Phénix and Super Phénix. One pin contains 9 fuel pellets with an average diameter of 5.37 mm, the other a 61.01 mm stack of sphere-pac fuel. The fuel pellets had a total length of 55.67 mm and were held in place with a spring. Hafnium oxide pellets have been placed at both ends of the fuel stack in order to decrease power peaking at the fuel stack ends. The sphere-pac fuel is a composite bed of beads of two diameters, around 0.75 and 0.05 mm, to enhance the packing fraction, and is also flanked by hafnium oxide pellets. Fig. 3 shows a schematic view of the two fuel concepts tested in MARINE.

Each pin is filled with a mixture of 99% He and 1% Ne. The presence of the 1% neon is essential to determine the He/Ne fraction accuracy after irradiation (higher precision than measuring the total He quantity). An overview of the fuel/targets irradiated in MARINE is given in Tables 1 and 2.

Two fabrication routes were followed by JRC-ITU (Fernández et al., 2002; Pouchon et al., 2012) to realise the two types of fuel (pellet and sphere-pac):

Pellets (see Fig. 4). The synthesis of the fuel pellets was performed by a combination of sol gel and infiltration methods. It required a number of steps including:

- Production of porous UO_2 beads (without americium) by the sol gel external gelation route to give beads with a polydisperse size distribution.
- Infiltration of the porous beads with americium solution and subsequent calcination.
- Pressing of the beads.
- Sintering of the compacted material.
- Control and selection.

Sphere-pac (see Fig. 5). The synthesis of the fuel fractions for the particle fuel was performed by group conversion from mixed nitrate solution, which was prepared by mixing Uranium with Americium nitrate solution in the required ratio. It required a number of steps including:

- Preparation of small size fraction by sol-gel external gelation route.
- Preparation of large size fraction by the sol gel external gelation route to yield monodisperse sized particles.
- Sintering of the beads.
- Control and selection.

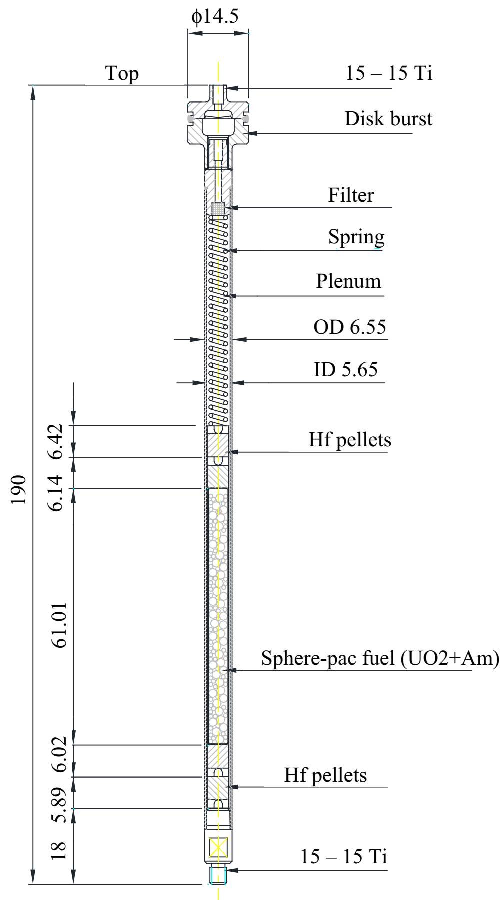

Fig. 2. Layout of the fuel pins for the MARINE irradiation (left side sphere-pac, right side pellets).

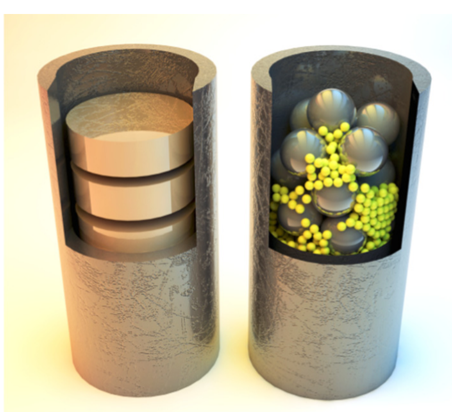

Fig. 3. Pellet versus sphere-pac concept.

Representative pictures of fuel pellets and spheres are presented in Figs. 4 and 5. Fig. 5 shows the sphere-pac fuel in the two sizes selected: 750 μm (top), 50 μm (bottom).

The sphere-pac fuels were particularly challenging to prepare, and the density was less than the initial design values, despite a concerted campaign of pre-tests to optimise the conditions of production. The pins are piled with the two separate fuel stacks placed in the highest flux position (i.e. close to the centre of the core).

3. Experimental

The MARINE irradiation is carried out in a channel of a TRIO 131 rig with a standard rig head (see Fig. 6). This is one of the standard re-usable irradiation rigs employed in the HFR.

The in-pile section holding the pins consists of three elements:

- The sample holder containing the two pins which constitute the 1st containment for the irradiation experiment.

Table 1
Composition of the fuel irradiated in the MARINE experiment.

Pin Nr.	Composition	Isotopic composition	Fuel density [g cm^{-3}]	Fuel volume [cm^3]	Free volume [cm^3]	^{241}Am contents [g]	^{238}U contents [g]	^{235}U contents [g]
1 Spheres	$U_{0.87}Am_{0.13}O_{1.935}$	$UO_2 + {}^{241}Am$	6.13[a]	1.530	2.47	1.08	7.152	0.051
2 Pellets	$U_{0.86}Am_{0.14}O_{1.93}$	$UO_2 + {}^{241}Am$	9.45[a] 10.45[b]	1.261	2.74	1.58	10.017	0.072

[a] This overall density takes into account both the density of the spheres and the overall packing density of the 2 size fractions in the sphere-pac column as well as the gas gap between the pellet and the cladding in the pellets column. This density, often called smear density, is given to be able to compare directly Sphere-pac with pellet fuel.
[b] This density represents the real density of the fuel (only the pellets) as fabricated.

Table 2
Isotopic vector of uranium.

Isotope	Abundance (%)
U-234	0.0041
U-235	0.7116
U-236	0.001
U-238	99.2838

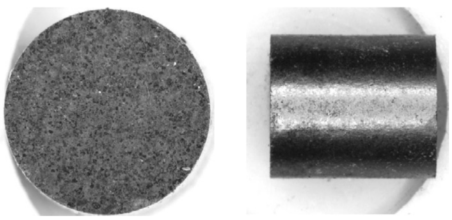

Fig. 4. Am-bearing blanket pellets fabricated for the MARINE irradiation.

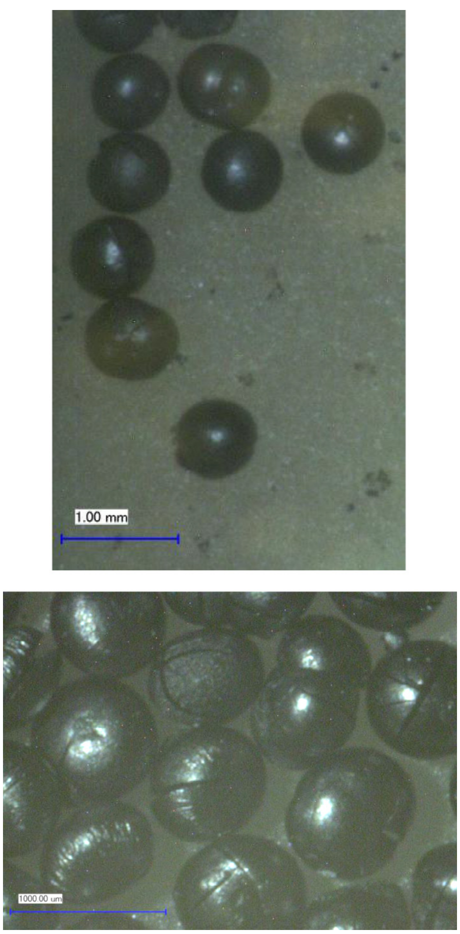

Fig. 5. Am-bearing blanket spheres for the MARINE irradiation in two size fractions: small (≈750 μm, top picture) and large one (≈50 μm, bottom picture).

- The TRIO-131 channels contains the sample holder which constitute the 2nd containment for the irradiation experiment.
- The rig head, representing the transition between the in-pile section and out-of-pile installation, carries all instrumentation leads and gas tubes.

The other two TRIO-131 channels are filled with dummy sample holders made of aluminium. Table 3 shows the radial dimensions of the TRIO-131 channels and the sample holder:

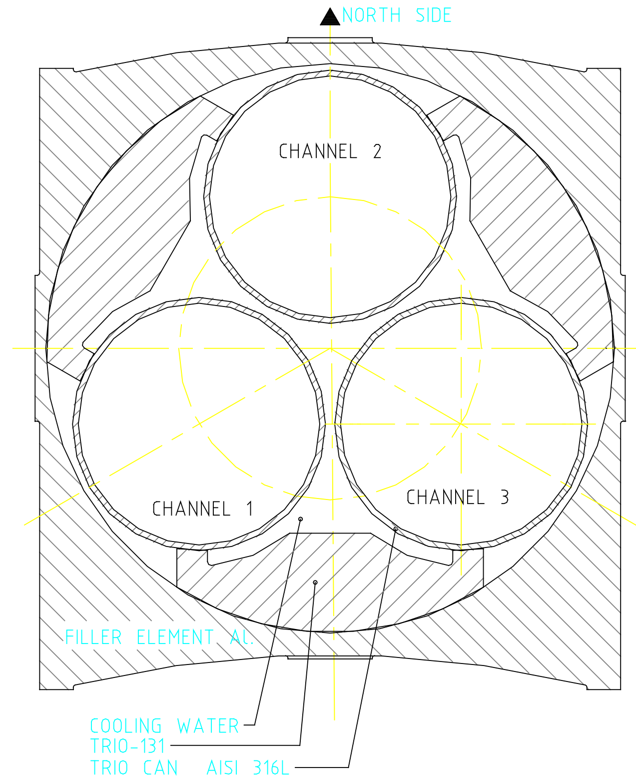

Fig. 6. TRIO-131 capsule for experiment (North orientation on top). Channel 2 (top position) is the channel employed for MARINE.

In order to reach the required average temperature in each pin, the radial gas gap between the 1st and the 2nd containments has been axially tailored (see Table 4).

The sample holder consists of three main sections (see Fig. 7). The scientifically most relevant part is the lower section where the experiment is located.

More in details the lower section comprises:

- A tube of AISI 321 containing the irradiation experiment and the pressure transducers.
- An open tube made of TZM (a Molybdenum alloy), the shroud, containing pins 1 and 2. The TZM shroud incorporates also some scientific instrumentation, such as thermocouples and neutron fluence detector sets.
- The sealed pins of 15–15 Ti steel cladding tubes with the fuel pellets and sphere-pac column.

Both pins, as well as the TZM shroud, are immersed in a Na bath for enhanced thermal bonding and concomitants heat removal. The Na fills the 1st containment and it is in contact with the shroud, with the fuel pins and with the AISI 321 sample holder containing the shroud and the pins. See Fig. 8 for a schematic view and Figs. 9 and 10 for a picture of the pins and the sample holder.

A gas gap between the sample holder and TRIO channel is used to tune the temperature of the fuel and to create a barrier between the Na and the water of the cooling system.

The Na remains liquid during operation to improve the heating transfer and avoiding solid formation (too cold working temperature) or sodium boiling (too hot working temperature). The temperature above and just below the Na surface will be monitored by four dedicated thermocouples. In order to prevent oxidation of the Na, the plenum of the 1st containment is filled with high-purity He at 0.1 MPa and sealed after final assembly. The heat generated by fission and gamma absorption in the materials will be

Table 3
Main dimensions of sample holder and TRIO channels (at room temperature).

Part	Outer ∅ [mm]	Inner ∅ [mm]	Tube thickness [mm]	Material
TRIO channels	33.5	31.5	1	AISI 316
Sample holder	See Table 4	29.0	See Table 4[a]	AISI 321

[a] The outer dimension of the sample holder has been tailored to match the required fuel temperature.

Table 4
1st containment outer radius and resulting gas gap dimensions (at room temperature).

Axial location [m]		Diameter [m]	Radius [m]	Gap dimension [μm]
Bottom	−0.1877	0.03050	0.01525	500
−0.1877	−0.1275	0.03080	0.01540	350
−0.1275	−0.0718	0.03092	0.01546	290
−0.0718	−0.0120	0.03080	0.01540	350
−0.0120	0.04901	0.03090	0.01545	300
0.04901	0.1057	0.03080	0.01540	350
0.1057	0.2694	0.03050	0.01525	500
0.2694	0.3488	0.03110	0.01555	200
0.3488	0.3707	0.03080	0.01540	350
0.3707	0.4478	0.03050	0.01525	500
0.4478	Top	0.03110	0.01555	200

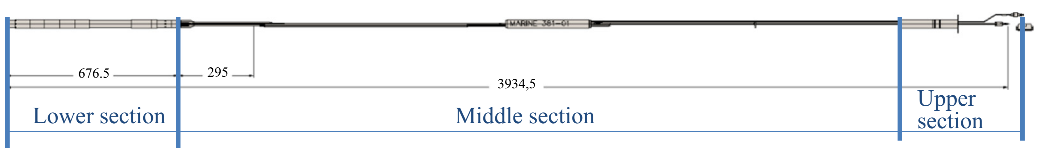

Fig. 7. Technical drawing of the sample holder.

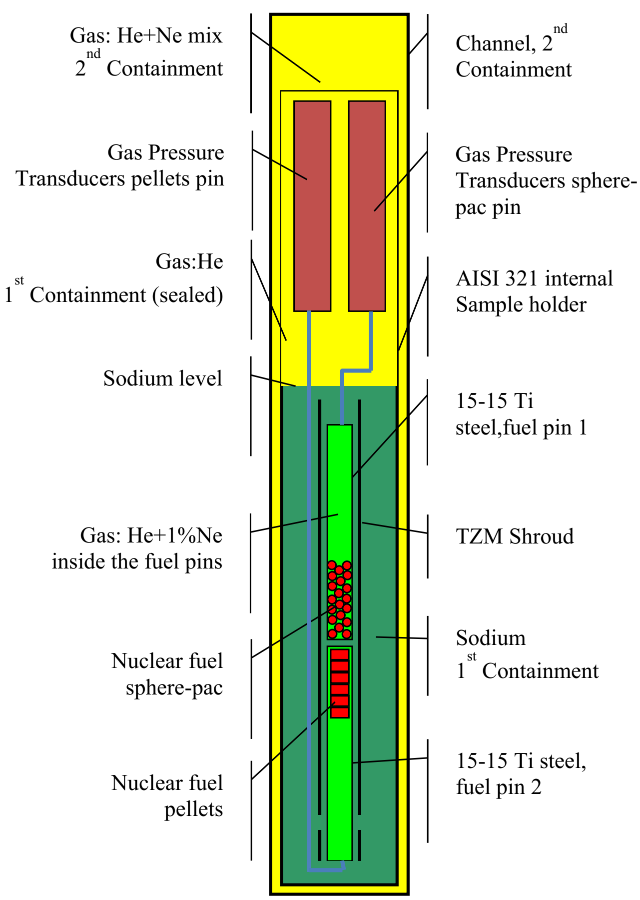

Fig. 8. Schematic view of the MARINE irradiation experiment.

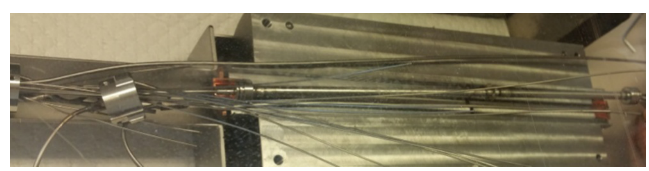

Fig. 9. The two fuel pins during the assembly.

radially dissipated through the Na bath, the structural materials and the gas gaps by conduction and radiation to the downstream primary coolant of the TRIO channel. The build-up pressure of the pins during the irradiation will be monitored on-line with two pressure transducers located above the pins and connected, via capillary tubes, to the fuel pins (see Fig. 8). The pressure transducers employed are based on Linear Voltage Differential Transformer (LVDT) technology which has already proven a very good reliability in the past.

The vertical position of the sample holder may be adjusted by means of a remotely operated vertical displacement unit (VDU) to optimise the sample holder position with respect to the neutron flux buckling in the HFR core. This allows the aligning of the two fuel stacks in the flux field to produce roughly the same linear power. The temperature in the sample holder can be adjusted by changing the gas mixture in the gap between the two safety containments. While the gap is initially filled with 100% Ne, the increase in heat generation due to the production of Pu by transmutation during irradiation can be compensated with He addition (Ne has a lower thermal conductivity than He and thus a better thermal insulation).

The scientific instrumentation of the MARINE experiment consists of 20 thermocouples (TCs) and 6 fluence detector sets (FD), 3 per pin. The fluence detector sets contain: a nickel-cobalt wire (1% Co), an iron wire, a titanium wire and a niobium wire.

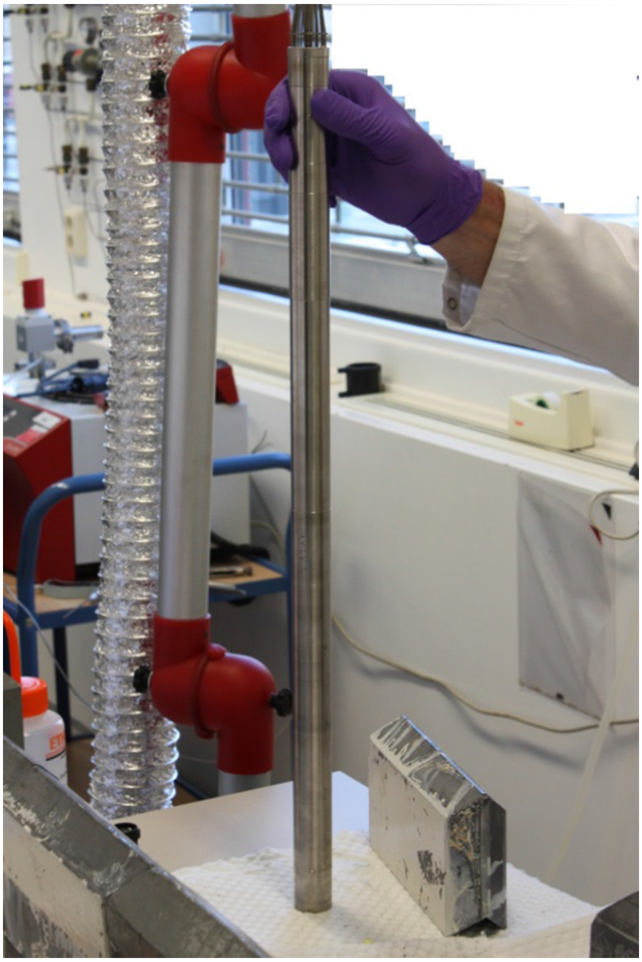

Fig. 10. The sample holder assembled.

The TCs used in the experiment are: 20 Type K thermocouples produced by Thermocoax to measure the temperature field at the Na level and above the Na level, the temperature of the pressure transducers as well as the cladding temperature of the pins. These TCs feature MgO as insulation material and AISI 304 L as sheath material. Their operating temperatures range from room temperature up to 800 °C. Thirteen TCs are located inside grooves milled along the outer diameter of the TZM tube. Four TCs are positioned at different levels to monitor the Na level and to detect leakage. Three TCs are located at different height along the pressure transducers.

Fig. 11 shows the pressure transducers and the TCs. Fig. 12 shows the TCs (bottom part of the picture) ready to be assembled in the TZM shroud (top part of the picture, the two half of a cylinder). In Fig. 12 the holder (in aluminium) is shown where the pressure transducers are inserted.

One of the challenges of the design of this experiment was the fact that the pins were fabricated at ITU and could not be sealed there because the final assembly, to connect the pressure transducers, have to be finalised in Petten. In contrast, traveling and handling unsealed pins containing americium can be an issue for any workshop. On top of this, the final assembly, where the connection capillary tubes have to be welded to the pressure transducers and to the pins, has to be done in a safe manner in a non-contaminated environment and outside a glove box. All the issues explained above have been solved with an innovative solution, using burst disks. These are safety devices, usually employed as safety relief valves. In case of overpressures in pressure vessels (see Fig. 13) they break immediately and cleanly as soon as the internal pressure reaches a certain value.

Fig. 11. The two pressure transducers and the thermocouples.

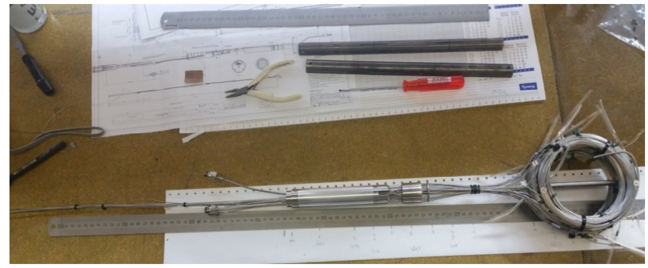

Fig. 12. Thermocouples (bottom part of the picture) ready to be assembled in the TZM shroud (top part of the picture).

Therefore, after the pins were filled with the fuels, already at ITU, they have been sealed by welding a burst disk component on top of each, to close the pins, see Fig. 9.

Doing so, the pin was perfectly sealed and ready to travel from Germany to the Netherlands. Then, after having welded the connection capillary tubes to the pressure transducers and to the pins, using a T-junction (see Fig. 14), the burst disks have been broken by pressurising the system at 0.8 MPa. After, the plenums of each pin have been filled with He (and 1% of Ne) and sealed again thereby closing the T-junction.

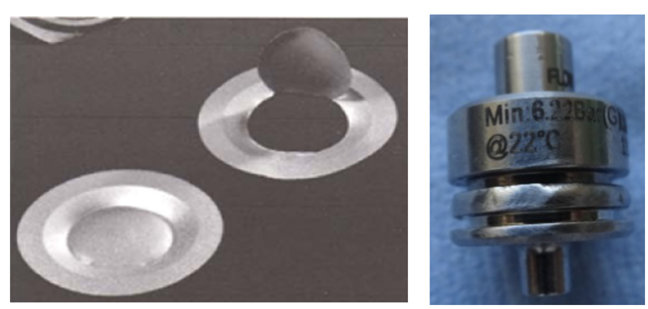

Fig. 13. The burst disks on the left and within its holder on the right.

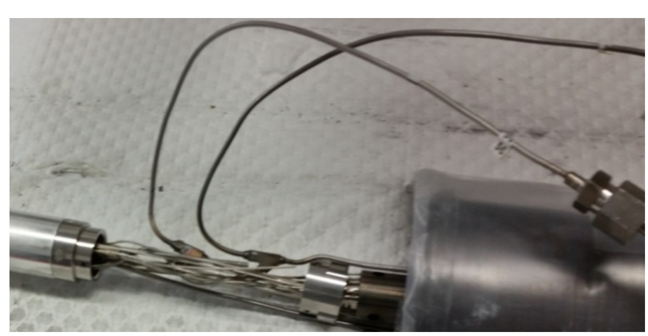

Fig. 14. A view of the T-junction used to pressurize the pins and break the burst disks.

4. Nuclear and thermal design

4.1. Neutronic assessment

The neutronic calculations have been performed by the Monte Carlo N- Particle code MCNP-4C3 (Briesmeister, 2000; Hendricks, 2001), using a representative HFR core model. The models for the TRIO containing MARINE was placed in this core model in position H6 north orientation (as shown in Figs. 15 and 16).

The burn-up evolution of the two pins of MARINE was calculated with the OCTOPUS code package (Oppe and Kuijper, 2004), which alternates spectrum calculation (using MCNP) and depletion calculation (using FISPACT-2007 Forrest, 2007).

Fig. 17 plots the expected power history (fission + neutron and gamma heating) of the MARINE pellets and sphere-pac fuel. It shows an increasing linear power with burn-up due to the production of Pu mainly from the reaction chain of Am.

The nuclear analyses have been performed for a total effective irradiation time of 15 cycles (450 full power days). The analyses have been performed introducing outages of 30 days every 90 days of irradiation, therefore, the cumulated period is 19 cycles (570 days). The outage period has been introduced to reflect the real working condition of the HFR. Indeed, there will be occasional stops due to maintenance of HFR during the MARINE irradiation.

The position-averaged fluence rates calculated by MCNP have been collapsed to the OSCAR3 7-group structure, which is applied

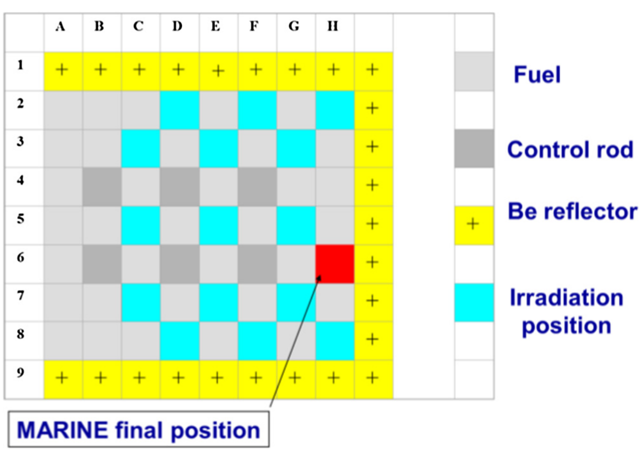

Fig. 15. HFR layout with the MARINE experiment loaded in H6.

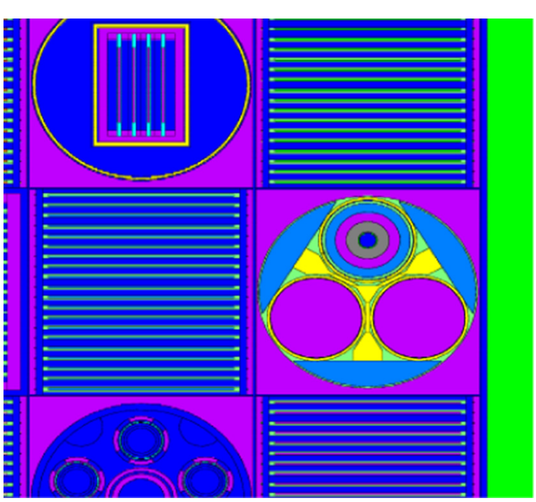

Fig. 16. A detail of the MCNP model of the experiment located in H6.

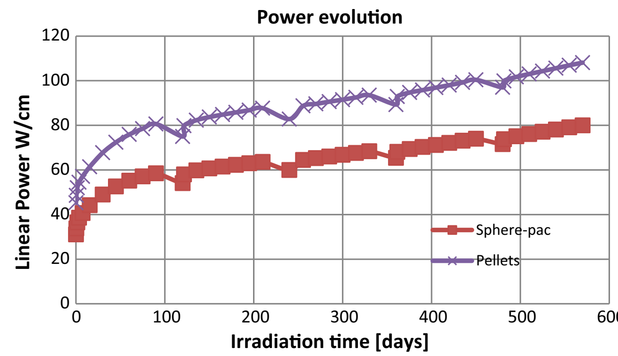

Fig. 17. Predicted power history of the Am-MABB fuel based on MCNP/FISPACT burn-up calculations.

for the HFR cycle calculations. The results are summarised in Table 5. All flux values are averaged over the fuel height (±0.300 m from the core centre-line). The statistical uncertainty from neutronic analyses (standard deviation, 1σ) is in the order of 1–2%.

One of the objectives of the MARINE experiment is to study the role of fuel microstructure and irradiation temperature on gas release. The aim is to reach a minimum He production of 2.7 mg cm^{-3} which is 2/3 of the expected He production of 2000 EFPD in the blanket of a SFR (typically 4 mg cm^{-3}) (Valentin et al., 2009). The calculated He production in Fig. 18, shows the evolution of the helium production in the 2 pins during irradiation. The production of helium, was calculated during the depletion calculation using FISPACT and adopting the same time frame assumed for the calculation of the power i.e. introducing outage of 30 days every 90 days of irradiation. However, due to the decay of ^{242}Cm (see Fig. 1), the production of helium is continuous also during the 30 days of outage as clearly visible in Fig. 18.

As shown in Fig. 18, the target objective of 2.7 mg cm^{-3} of He production should be reached after 336 days of irradiation in the HFR having therefore an acceleration rate of ≈4 (i.e. (2/3 2000)/336 ≈ 4). The acceleration of the He production in HFR compared to fast reactor is due to the high cross section of ^{241}Am in thermal neutron flux that leads to a speed-up of transmutation phenomena. In Table 6 the maximum plenum pressures at the end of the irradiation are conservatively estimated on the basis of full release of He and fission gases in the plenum volume at an estimated average plenum temperature (900 K). The column with the maximum pressure in Table 6 includes also the initial pressure due to the presence of the helium filled during assembly of the pins (0.1 MPa at 298 K results in 0.302 MPa at 900 K). The maximum pressure expected in the pin containing the pellets is 9.67 MPa.

Note that in order to have a similar pressure in both pins, they have a different length (plenum volume) because the pin at the bottom (shortest pin) require a longer capillary tube to be connected to the pressure transducers located in the upper part of the sample holder (see Figs. 2 and 8).

Table 5
Position averaged fluence rates for MARINE.

	Energy boundaries (eV)	Fluence rates [10^{18} m^{-2} s^{-1}]
Group 1	$8.208 \cdot 10^5/1.96 \cdot 10^7$	0.50
Group 2	$5.53 \cdot 10^3/8.208 \cdot 10^5$	0.69
Group 3	$4.00/5.53 \cdot 10^3$	0.51
Group 4	0.625/4.00	0.12
Group 5	0.248/0.625	0.07
Group 6	0.058/0.248	0.24
Group 7	0.0001/0.058	0.27
Total		2.40

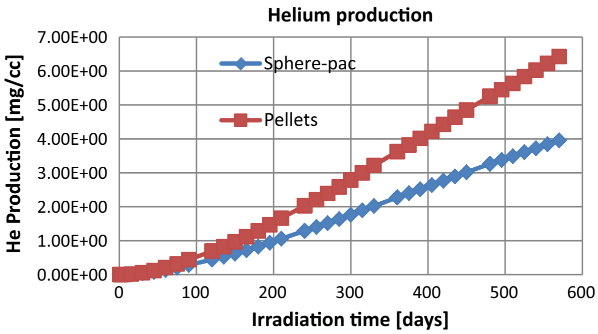

Fig. 18. Predicted helium production of the Am-MABB fuel based on MCNP/FISPACT burn-up calculations.

Table 6
Estimated maximum plenum pressure in the fuel pins.

Pins Nr.	Total gas (FP + He) [mol]	Temperature [K]	Plenum volume [cm^3]	Maximum pressure at 100% release [MPa]
1 (Sphere-pac)	1.89E−3	900	2.13	6.72
2 (pellets)	2.48E−3	900	1.99	9.67

4.2. Thermo-mechanical assessment

In order to understand the behaviour of the experiment, thermal analyses for the beginning (BOI) and the end of irradiation (EOI) have been performed. The cases reported refer to BOI and to the design EOI (i.e. after 420 days of irradiation, 15 cycles). The design allows tuning the temperature by changing the gas mixture in the gap between the sample holder and the TRIO

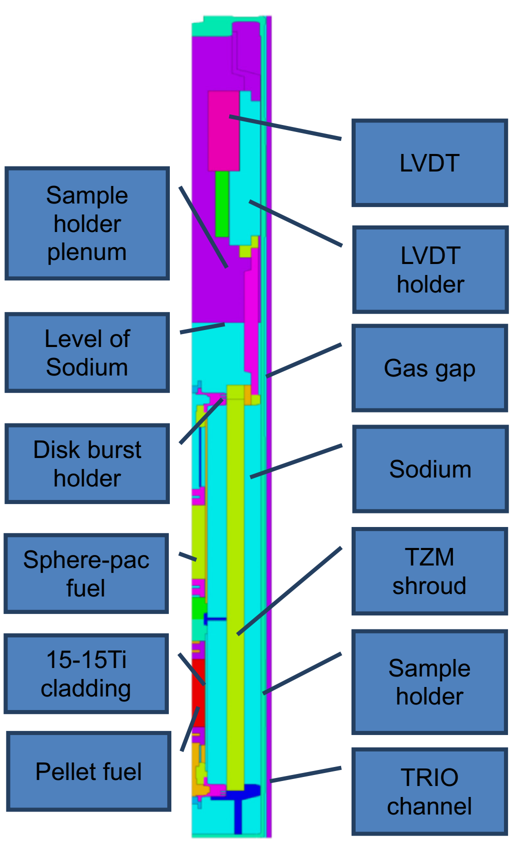

Fig. 19. Axisymmetric FE model of MARINE.

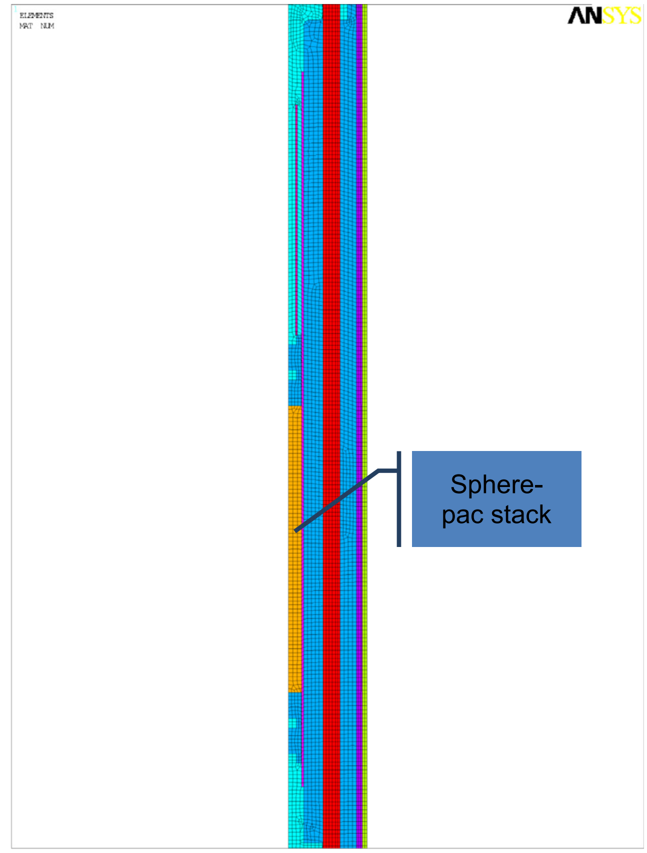

Fig. 20. Mesh distribution at sphere-pac location.

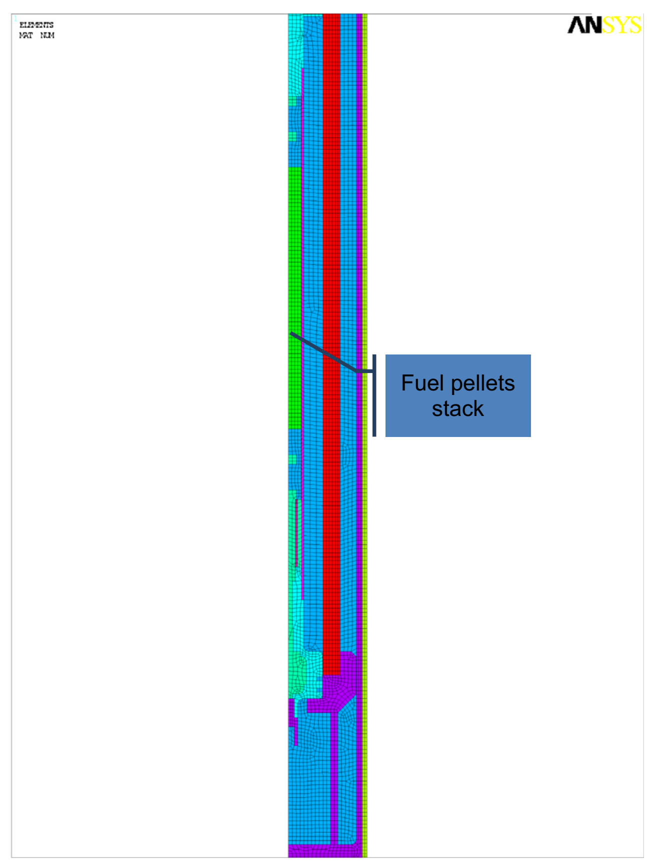

Fig. 21. Mesh distribution at pellet location.

channel. To determine the safety margin, the analyses were repeated with the gas gap completely filled with He or Ne.

For these calculations the FEM code ANSYS version 14.0 was used. To optimise the dimensions, an axisymmetric finite element model of the MARINE experiment was prepared.

The temperatures and stresses were calculated using a single axisymmetric model coupling a thermal and a mechanical analysis. The model used can be seen in Fig. 19.

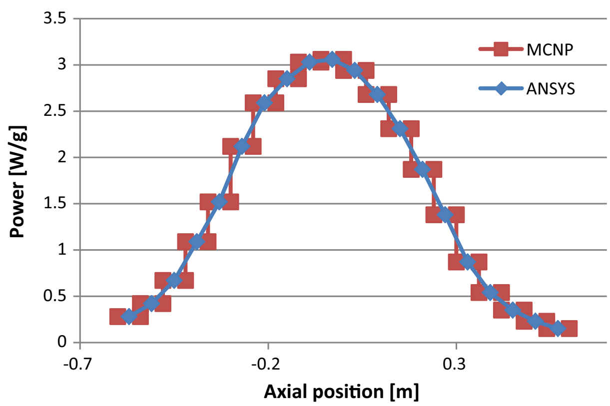

Fig. 22. Nuclear heating (gamma + neutron) for position H6.

The different colours shown in Fig. 19 refer to the different materials used in the analysis. The mesh is too dense in the axial direction to be shown in a single image. Therefore, Figs. 20 and 21 show the mesh distribution at the sphere-pac and at the pellet fuel location.

The nuclear heating (gamma + neutron) in position H6 depends on the axial position and is based on MCNP calculations. Fig. 22 shows the heating profile resulting from these MCNP calculations with a 6th order polynomial regression.

In the fuel pellets, besides nuclear heating, additional heat is generated due to fission heating.

The fission average linear heating rate for the sphere-pac is 31.1 W cm^{-1} at the beginning of irradiation and 80 W cm^{-1} at the end of the 15-cycle irradiation, if the experiment stays in the chosen HFR position. For the fuel pellets the fission average heating is 45.3 W cm^{-1} at the beginning of irradiation and 108.1 W cm^{-1} at the end of irradiation.

The model was verified for radial heat transfer by comparison with a 1D model.

Four design cases have been considered labelled:

- case 1 BOI-helium;
- case 2 EOI-helium;
- case 3 BOI-neon;
- case 4 EOI-neon.

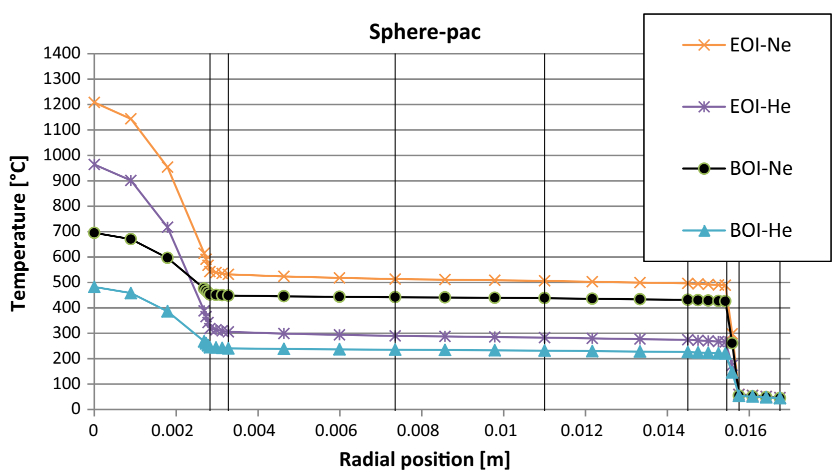

Fig. 23. Comparison of radial temperature distribution for the different load cases, sphere-pac.

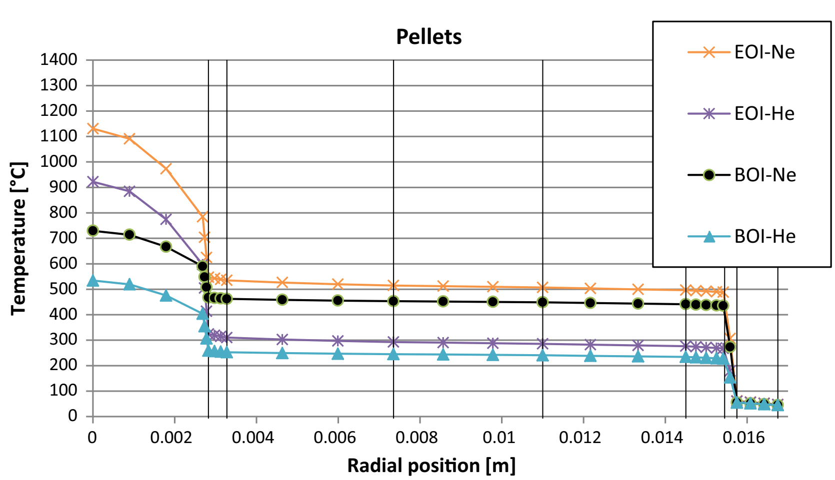

Fig. 24. Comparison of radial temperature distribution for the different load cases, fuel pellets.

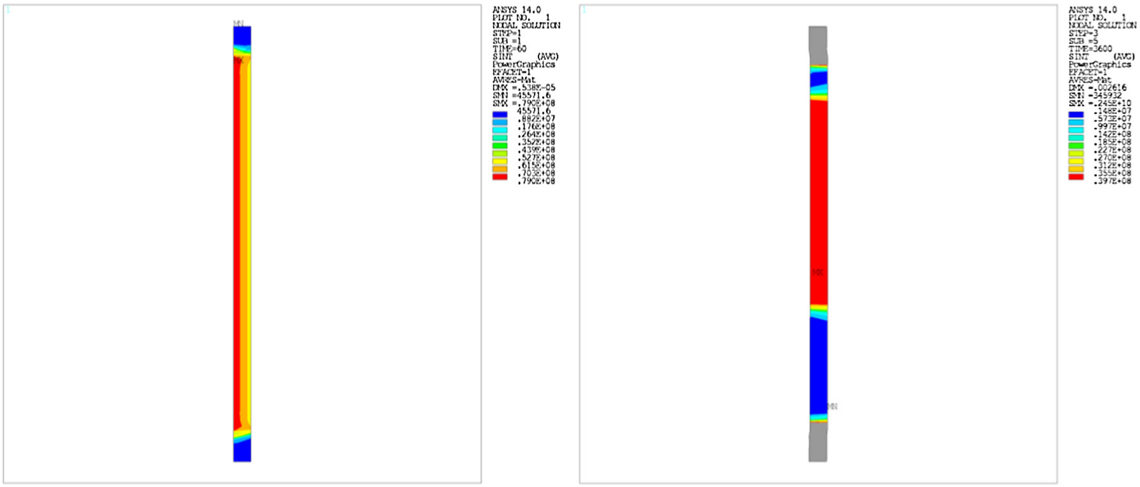

Fig. 25. Equivalent intensity primary (left) and secondary (right) stress in pin 2 (pellets) [Pa].

All cases describe the situation at a certain time in point (Beginning or End of Irradiation) when it is used 100% He or Ne in the gas gap between the 1st and 2nd containment.

Figs. 23 and 24 show the radial distribution of the temperature for the sphere-pac and pellet fuel, which are possible to reach during the irradiation by changing the gas mixture in the gas gap.

From the figures above, it is clearly visible that the model predicts a maximum fuel temperature of 1200 °C in the absence of fuel restructuring which at this temperature is not expected. The temperatures of the cladding will always remain well below 650 °C which is considered a safe limit for 15–15 Ti steel. The pressure in the pins will increase during the irradiation due to the production of gas (fission product and helium). As calculated in the previous section, the maximum pressure reached from the pin contained pellets is 9.67 MPa. In reality, such pressure will never be reached because the assumption that the gases will be fully released from the fuel is too conservative. Anyway, for safety reasons, the claddings have been assessed for a pressure of 10.0 MPa. The stress assessment is according to ASME Boiler and Pressure Vessel Code, Section III.

The stresses in the pins consist of the primary stress caused by an assumed 10.0 MPa internal pressure for pin 2 and secondary stresses caused by thermal loading. Fig. 25 shows the primary (on the left side) and the secondary (on the right side) stress (stress intensity) in pin 2. The primary stress (values averaged on the thickness of the cladding) in pin 2, is about 70 MPa while the secondary stress is about 39 MPa.

5. Conclusions

Minor Actinide bearing blanket Fuel (MABB) is an option to burn minor actinides heterogeneously outside the core of a sodium-cooled fast reactor. The results of the MARINE experiment will provide a better understanding and assessment of the comparative irradiation behaviour of sphere-pac and pelletized fuel with respect to helium release/trapping and to fuel swelling and fuel-cladding interaction. Moreover, MARINE will provide essential information on gas release continuously during irradiation because of the presence of the pressure transducers that will monitor the pressure of the pins online.

The irradiation will reach the target of an helium production of 2.7 mg cm^{-3} after 12 cycles (as seen in the neutronic calculation in Section 4.1). At the end of the experiment, the analysis of the fluence detector sets will confirm the exact neutron fluence reached during the experiment. Then, post-irradiation neutronic simulations, to reassess total helium production, will probably be performed within the JPNM framework inside a so called Pilot Project. These values will be compared with the values obtained by post-irradiation gas puncturing (to determine the released He) + annealing test (to measure He retained in the fuel). A comparison of this data (also with the data of the pressure of the pins collected online) will confirm the amount of helium produced and released which will be then correlated with the fuel temperature, thus enabling a direct performance comparison between pellet and sphere-pac fuel.

Other investigations will be probably performed during the Post Irradiation Examination (PIE): profilometry, gamma-scanning and spectrometry, sample metrology, ceramography as well as Scanning Electron Microscopy and Electron Micro-Probe analysis to determine the microstructure evolution during irradiation. One of the expected advantages of the sphere-pac geometry compared with pellets is to be able to better accommodate the swelling of the fuel. Therefore, it will certainly be part of the PIE to assess any interaction between fuel and cladding.

Acknowledgments

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