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Next generation fuel irradiation capability in the High Flux Reactor Petten

Michael A. Fütterer [a,*], Elio D'Agata [a], Mathias Laurie [a], Alain Marmier [a], Francesco Scaffidi-Argentina [a], Philippe Raison [b], Klaas Bakker [c], Sander de Groot [c], Frodo Klaassen [c]

[a] European Commission, Joint Research Centre, Institute for Energy (JRC-IE), P.O. Box 2, NL-1755 ZG Petten, The Netherlands
[b] European Commission, Joint Research Centre, Institute for Transuranium Elements (JRC-ITU), D-76334 Eggenstein-Leopoldshafen, Germany
[c] Nuclear Research and consultancy Group (NRG), P.O. Box 25, NL-1755 ZG Petten, The Netherlands

A B S T R A C T

This paper describes selected equipment and expertise on fuel irradiation testing at the High Flux Reactor (HFR) in Petten, The Netherlands. The reactor went critical in 1961 and holds an operating license up to at least 2015. While HFR has initially focused on Light Water Reactor fuel and materials, it also played a decisive role since the 1970s in the German High Temperature Reactor (HTR) development program. A variety of tests related to fast reactor development in Europe were carried out for next generation fuel and materials, in particular for Very High Temperature Reactor (V/HTR) fuel, fuel for closed fuel cycles (U–Pu and Th–U fuel cycle) and transmutation, as well as for other innovative fuel types. The HFR constitutes a significant European infrastructure tool for the development of next generation reactors. Experimental facilities addressed include V/HTR fuel tests, a coated particle irradiation rig, and tests on fast reactor, transmutation and thorium fuel. The rationales for these tests are given, results are provided and further work is outlined.

1. Introduction

The High Flux Reactor Petten (HFR) is a valuable European infrastructure tool for testing of fuels and materials for next generation nuclear reactors. Depending on the size of the irradiation target, several highly standardized rig types can be used allowing for up to four thimbles to be irradiated simultaneously in a single irradiation position of which 17 are available. These rigs are externally cooled by HFR primary cooling water. The vertical position of individual thimbles can be remotely adjusted to center the target in the maximum flux position. The rigs can be rotated cycle to cycle to flatten radial fluence gradients. Temperatures are adjusted by suitable helium/neon blends. Wiring for instrumentation is also provided. Rigs are equipped with a top rig head containing valves, connectors and electric drives. Umbilical cords on top of the rig head route gas lines to gas supply and off-gas installations, and instrumentation cables to data acquisition systems. Rig heads and umbilical cords are pressurized with nitrogen to keep them dry. Specially designed filler elements in the irradiation position allow flux tailoring by variation of the cooling water volume around the rig.

2. V/HTR fuel element irradiations

As part of ongoing European V/HTR R&D and based on more than 20 years of HFR experience related to the former German HTR program, new irradiation campaigns were started in 2004–2005 (HFR-EU1bis [1]) and 2006–2008 (HFR-EU1) with the objective to better define the performance limits of existing classical HTR fuel pebbles from Germany and China with respect to acceptable operating temperature and burn-up. Both parameters are fundamental to the Generation IV VHTR concept. Qualification for low fission gas release in normal operating conditions and during simulated cooling accidents is a licensing requirement for newly developed HTR fuel. These tests and complementary post-irradiation examinations including burn-up measurements can be performed in JRC and NRG laboratories in Petten and Karlsruhe.

HTR fuel pebbles have a diameter of 6 cm. They contain 10–15 thousand coated particles homogeneously distributed in a graphite matrix. Each coated particle has a diameter of about 1 mm and comprises a UO_2 kernel successively coated with layers of carbon and silicon carbide to ensure mechanical stability and leak tightness against release of fission gases that build up during irradiation.

In HFR-EU1bis test, five AVR fuel pebbles from Germany were irradiated at a constant central temperature of 1250 °C, which is higher than in most previously performed experiments of this type and relevant for VHTR operating conditions. The sample holder structure is sketched in Fig. 1.

The individual pebbles were enclosed in graphite half shells for good thermal contact and the resulting cylindrical stack of graphite and fuel was inserted in a steel capsule. In-pile instrumentation included thermocouples, gamma scan wires and neutron fluence detector sets. This steel capsule was then inserted in a standard

* Corresponding author. Tel.: +31 224 565158; fax: +31 224 565627.
E-mail address: michael.fuetterer@jrc.nl (M.A. Fütterer).

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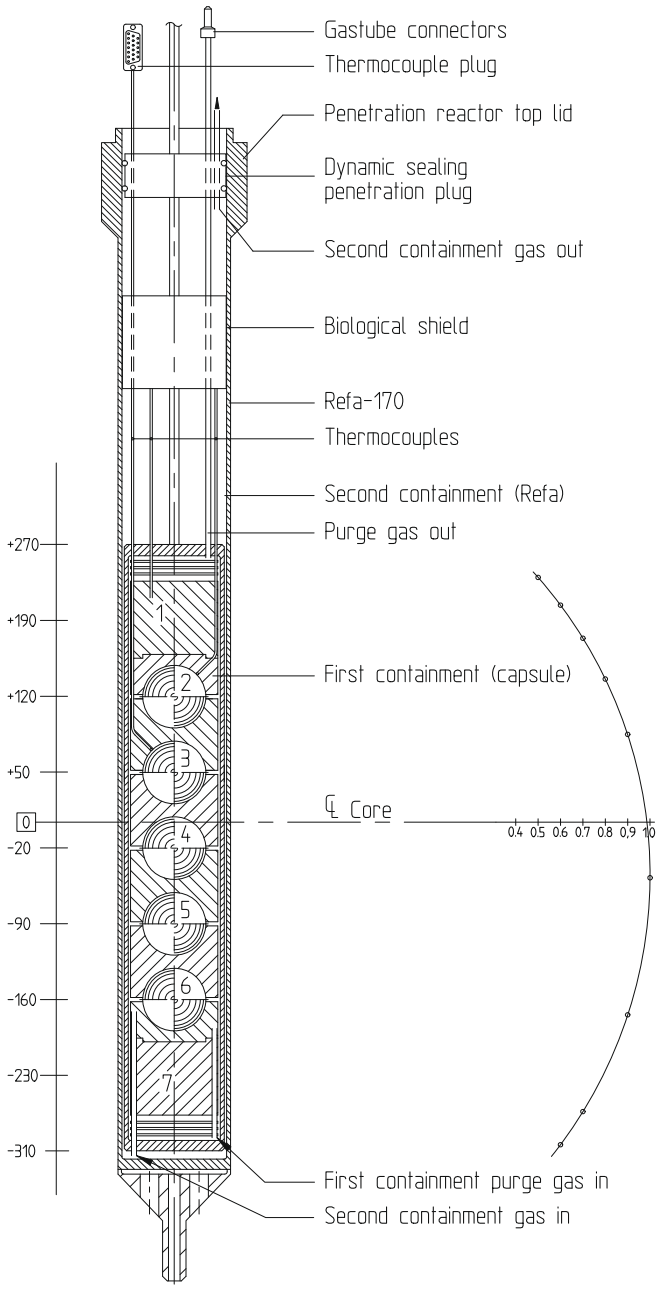

Fig. 1. HFR-EU1bis sample holder (axial scale in millimeter).

REFA-170 rig. The resulting gas gaps, graphite/steel capsule, and steel capsule/REFA-170 were tailored such that the resulting axial temperature profile was as flat as possible. Suitable blends of He and Ne in either gap were used for temperature adjustment.

Due to power depletion with burn-up, the desired constant central temperature implied a gradual rise in surface temperature. During 249 equivalent full power days (efpd), a maximum burn-up of 11% fissions per initial metal atoms (FIMA) was achieved, measured by gamma spectrometry. During the irradiation, gas samples were taken regularly and analyzed by gamma spectrometry for fission gas release. The ratio between measured release rate R and fission gas 'birth rate' B (R/B) was used as the indicator of fuel particle integrity and plotted against the burn-up as shown in Fig. 2.

Although the R/B fraction observed in HFR-EU1bis started about a factor 100 higher than in earlier experiments, the absolute release fraction measured for several isotopes was still very low and of the order of 3×10^{-6}. Only the measurements of Xe-133 were somewhat higher and erratic, and were attributed to decay of precursor fission products, in particular I-133. While modeling and interpretation work are ongoing, it is quite safe to postulate a manufacturing defect in one or several particles contained in the fuel pebbles. The often observed increase in R/B towards the end of irradiation may be attributed to U and Th contamination of the graphite (matrix and half shells). Neutron irradiation would progressively convert these impurities to fissile isotopes which then produce fission products released quite easily into the purge gas thus raising the R/B ratio.

Post-irradiation out-of-pile heating tests of the irradiated pebbles at JRC-ITU [2] are currently being completed and are expected to locate the source of the increased release.

While the HFR-EU1bis irradiation was completed in October 2005, the HFR-EU1 irradiation [3] started in September 2006 and has so far achieved a maximum burn-up of approximately 11% FIMA. The objective is a lower surface temperature (900–950 °C) but a higher burn-up than in HFR-EU1bis. Both AVR pebbles (the same type as in HFR-EU1bis) and Chinese pebbles for the HTR-10 reactor are being irradiated in two separate, stacked capsules.

Since the beginning of irradiation, the R/B of the AVR pebbles has steadily increased from an extremely low 10^{-8} to 10^{-7} at 11% FIMA, which may again be attributed to fission gases from impurities in the graphite. These low values are consistent with earlier experiments and hint at the absence of manufacturing defects.

Due to massive failure of thermocouples after more than a year of high temperature operation, the experiment is currently on hold for a safety assessment. It is likely that the original burn-up target of 20% FIMA cannot be achieved. To avoid a situation where instrumentation issues would limit the achievable burn-up, JRC is involved in the development and testing of new thermocouples with improved durability under irradiation at high temperature.

A further HTR fuel irradiation tests is being prepared with new fuel for the South African Pebble Bed Modular Reactor (PBMR). A complete design for an alternative HTR fuel type ('compacts') based on coated particles compacted with matrix graphite to a cylindrical rodlet is also available at the HFR.

3. V/HTR fuel particle irradiations

In addition to HTR fuel element irradiations, two irradiation experiments code-named PYCASSO-I and -II are under preparation to determine the effect of high temperature irradiation on thermomechanical properties of various coating materials for TRISO coated particle fuel. Partners involved in this irradiation are CEA (France), JAEA (Japan) and KAERI (South Korea), who have delivered dummy particles with different types of PyC/SiC coating combinations and ZrC. The partnership for PYCASSO was initiated by the European RAPHAEL project [4] and is integrated in the Generation IV International Forum VHTR Fuel and Fuel Cycle project.

These 'analytical' irradiations focus on determining the effects of neutron irradiation in the temperature range of 900–1100 °C, excluding effects due to the presence of fuel, such as pressurization or chemical attack by fission products. These irradiations can therefore be considered separate effect tests, dedicated to the validation and parameter determination for related codes, which are developed and used to predict coating behavior under high temperature irradiation conditions.

The specimens consist of kernels of inert material (alumina or zirconia) with different coatings, coating materials and coating combinations:

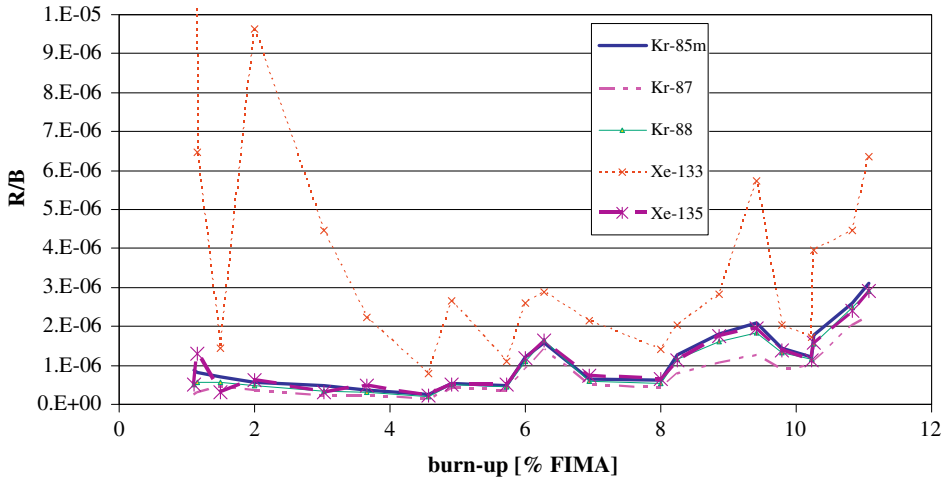

Fig. 2. HFR-EU1bis evolution of fractional fission gas release (*R*/*B*).

- CEA, France, has delivered alumina kernels with a number of different combinations of pyrocarbon and silicon carbide coatings.
- KAERI, South Korea, has delivered zirconia kernels with differently produced pyrocarbon layers.
- JAEA, Japan, has delivered alumina and zirconia kernels coated with a variety of silicon carbide, and zirconium carbide coatings in various combinations with pyrocarbon.

Key technical requirements are:

- Constant specimen temperature throughout irradiation.
- Accurate fluence determination for the samples.
- Minimum specimen activation for easy handling and post-irradiation examination, preferably in a glove box.
- Minimum chemical interaction, e.g., between samples and sample holder material.

These boundary conditions have led to the design of flat disk-like sample holders (diameter approximately 25 mm, cf. Fig. 3), where one layer of particles can be accommodated per sample holder in a hexagonal cavity. Sample holders are made of a high density, high thermal conductivity material, such that sufficient heat is generated through gamma radiation, while maintaining low thermal gradients in the sample holder.

The sample holder disks are then stacked into drums. The drums are thermally insulated radially by tailored gas gaps, and they are insulated axially to enable axial temperature steps between drums. A finite element model was applied to optimize the geometry of gas gaps and sample holder such that the temperature targets can be achieved (cf. Fig. 4), taking into account the non-uniform gamma radiation heating in the HFR core. The results of pre-experiment calculations including the gamma radiation heating curve are shown in Fig. 5.

The specimens are separated from the sample holder by a refractory metal (Mo alloy) foil. This foil will activate only weakly during irradiation and showed complete inertness towards sample holder material and specimens in high temperatures tests.

The PYCASSO-I irradiation started on 19 April 2008 with very good agreement between calculated temperature profile and measurements. PYCASSO-I and -II will be irradiated for approximately six and nine calendar months, respectively. Suitable post-irradiation examinations are currently being evaluated and will include high precision metrology and microstructural evaluation.

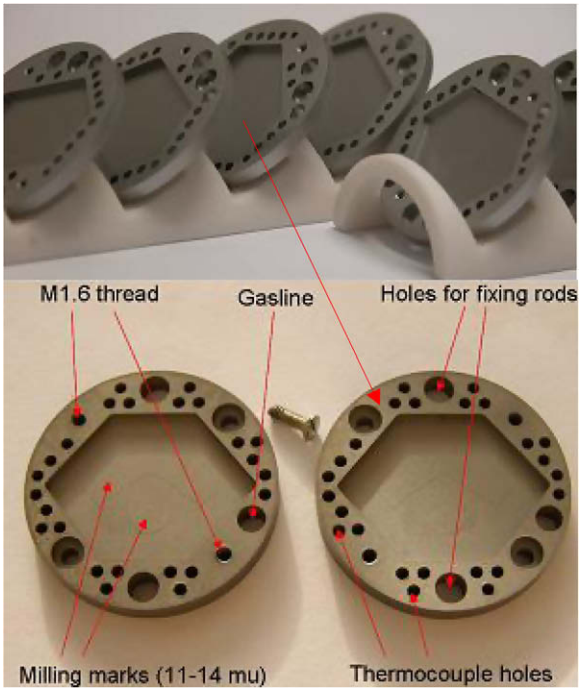

Fig. 3. Hexagonal sample holder cups for PYCASSO (outer diameter 25 mm).

4. Fast reactor fuel irradiations

Performance of liquid metal cooled fast breeder reactors with advanced specifications such as the Sodium Fast Reactor (SFR) or the Lead-cooled Fast Reactor (LFR), as well as feasibility of Accelerator Driven Systems (ADS) heavily depend on fuel integrity and proper fuel behavior. In the period between approximately 1970 and 1990 significant expertise was accumulated at the HFR [5] on testing and qualification of fast reactor fuels in three distinct test types. In recent years, this expertise was recovered.

4.1. Fast reactor and transmutation fuel base irradiations

Base irradiation of fast reactor fuel in the HFR is generally done by placing the fuel in stagnant sodium. Stagnant sodium is used to transfer the heat produced in the fuel pins to the surrounding cooling water. JRC-IE operates dedicated sodium filling equipment for

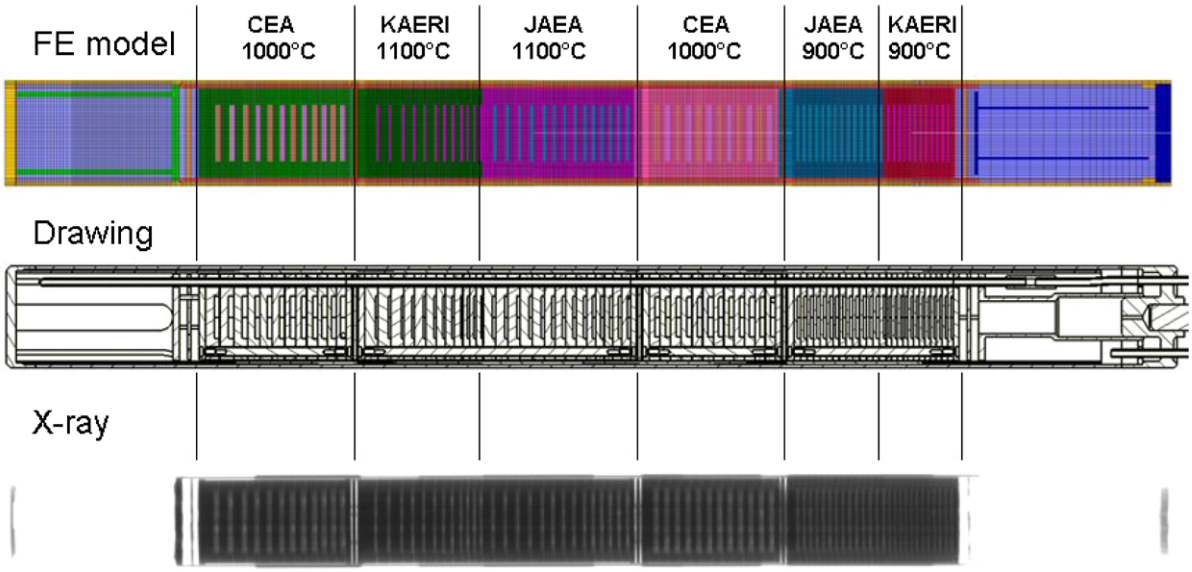

Fig. 4. PYCASSO-I FEM model and temperature regions (diameter 25 mm, length 700 mm).

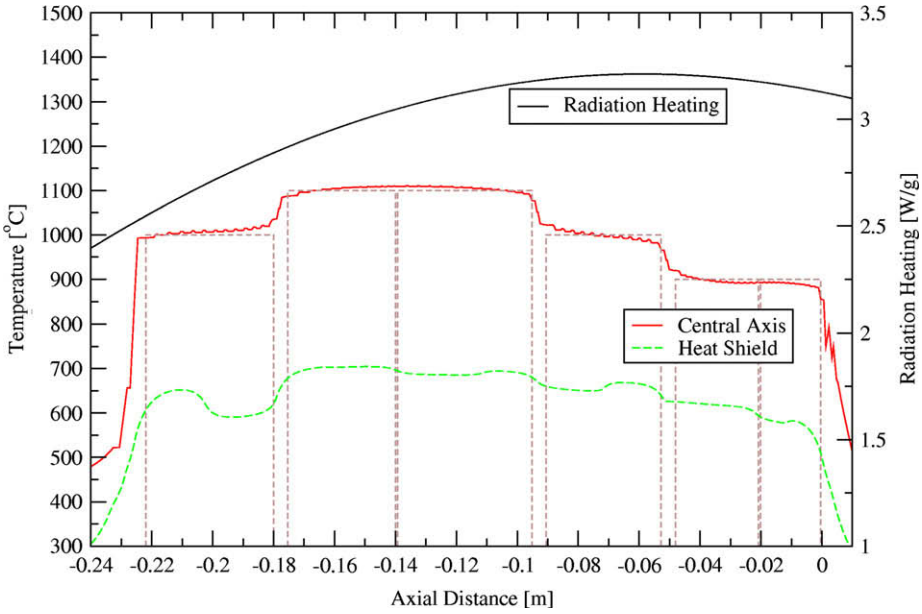

Fig. 5. PYCASSO-I pre-calculated temperature profile and radiation heating along central axis.

irradiation tests (cf. Fig. 6). Irradiations can be performed in the HFR with hafnium or cadmium screens so as to harden the neutron spectrum thus making irradiation conditions highly representative of fast reactors.

A recent example of such a fast reactor fuel base irradiation is Collaboration On Nitride Fuel IRradiation and Modeling (CONFIRM) whose purpose is to investigate the properties of uranium-free nitride fuels. The CONFIRM irradiation was prepared in 2006 and consists of testing two plutonium fuel pins with 30% plutonium–nitride in a zirconium–nitride inert matrix; $(Pu_{0.3}, Zr_{0.7})N$. In order to tune the neutron spectrum to typical conditions of fast reactors, the thermal part of the HFR spectrum is shielded by a hafnium shroud, which effectively absorbs thermal neutrons.

The CONFIRM fuel pins, fabricated at Paul Scherrer Institute in Switzerland, arrived in Petten in spring 2007. The sample holder was manufactured, assembled, and went through the commissioning and safety approval process. Thermocouples are monitoring the temperature of the pin cladding. They are held at approximately 500 °C, which corresponds to realistic fast reactor sodium temperatures. The CONFIRM irradiation was completed by July 2008. An example of the temperature reading during start-up is plotted in Fig. 7.

4.2. Fast reactor fuel ramp testing

High power, short duration fuel testing (ramp testing) in stagnant sodium can also be performed in the HFR. A recent example of this is the FUJI project in which the high power behavior of pellet, Sphere-Pac and VIPAC fuel was compared [6]. A horizontal cross section of the irradiation rig and the sample holders which are used for fast reactor fuel ramp testing is shown in Fig. 8.

Fig. 6. Sodium filling of the CONFIRM irradiation.

Two fuel pins can be placed next to each other, each fuel pin inside a sample holder. These two sample holders are placed in an irradiation rig. The fuel pin is surrounded by stagnant sodium. This stagnant sodium is surrounded by a molybdenum shroud which homogenizes temperatures and holds thermocouples. Outside the molybdenum shroud is another stagnant sodium layer followed by a dual stainless steel containment with a gas-filled gap in between. The temperature of the fuel pin can be controlled by changing the gas mixture between these two containments. Power is varied by moving the sample holder relative to the HFR core. The maximum acceptable power in this ramp test device is 700 W/cm per fuel pin. A steel absorber between the HFR core and the sample holders absorbs excess neutrons thus homogenizing circumferential power distribution in the fuel pins.

4.3. Fast reactor fuel accident simulation tests

Accident simulation tests were performed in the past in which fast reactor fuel pins were irradiated at extremely high linear power (up to 2000 W/cm) with sodium (Na) or sodium–potassium (NaK) cooling. Large power variations were obtained by a combination of moving the fuel test device towards the core and, in some designs, by removing BF_3-gas surrounding the fuel pin. These high powers led to extensive melting of the fuel pin. Also, loss of-coolant accidents were simulated on fast reactor fuel pins [7]. A 3D drawing of such an experiment is shown in Fig. 9. At the start of the experiment, the fuel pins are submerged in a liquid NaK bath, which effectively conducts the heat away from the fuel pins thus keeping them at desired temperatures. Using gas pressure, the liquid NaK can be transferred from the fuel into a reservoir thus interrupting cooling and simulating a loss-of-coolant accident. The last fast reactor fuel accident simulation test was performed

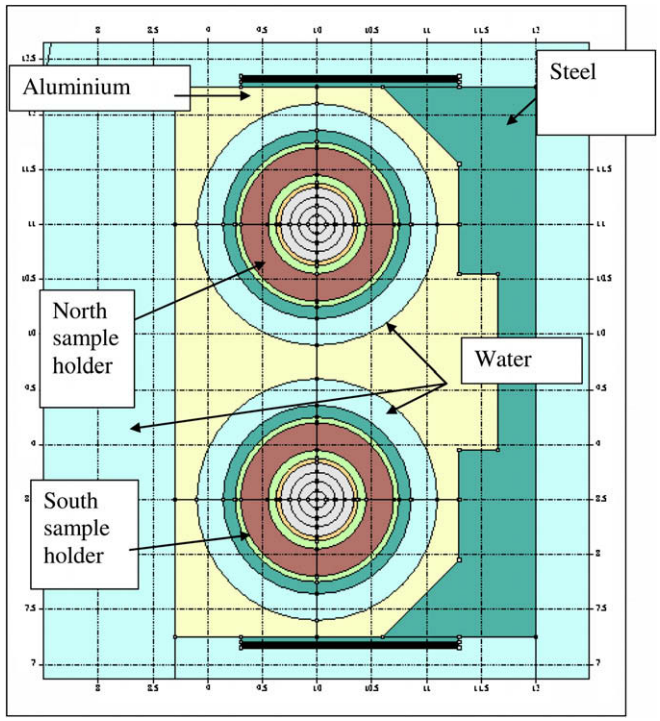

Fig. 8. Horizontal cross section (500 mm × 350 mm) of the sample holder for fast reactor fuel ramp testing.

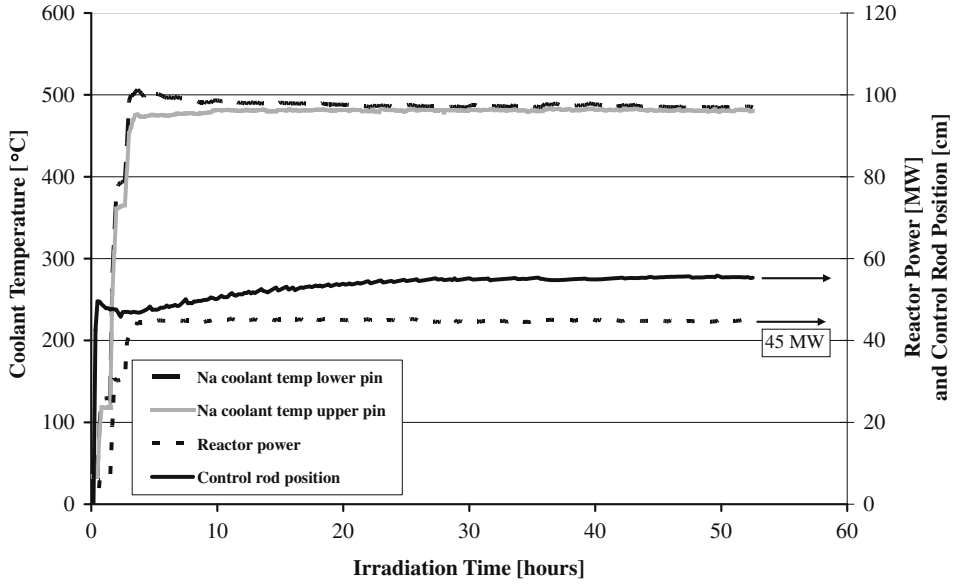

Fig. 7. Temperatures and reactor power 50 h into start-up of CONFIRM (25 November 2007).

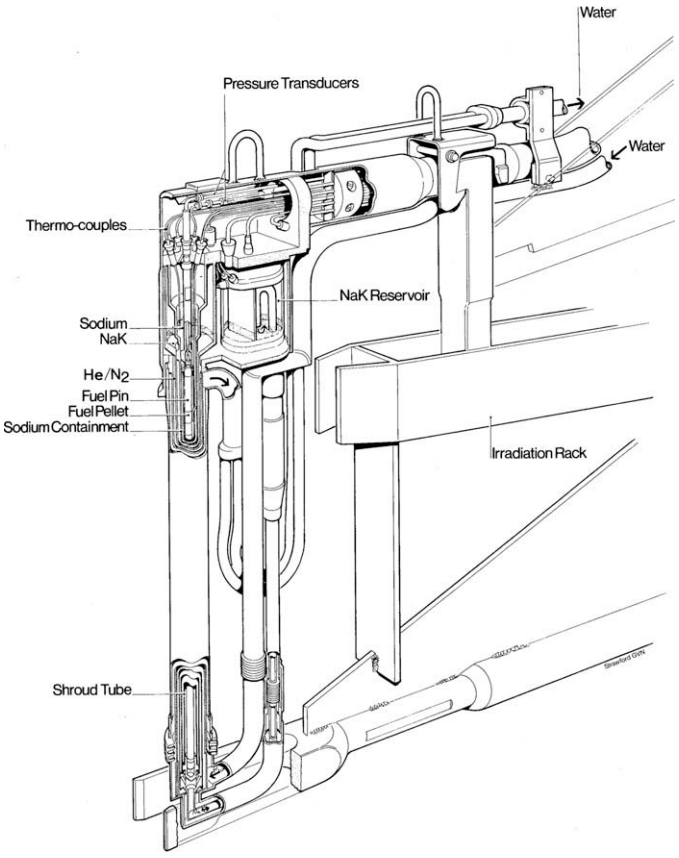

Fig. 9. 3D drawing of the LOC experiment (total height approximately 1 m).

in the HFR approximately 20 years ago. However, due to increasing momentum of fast reactor fuel development, NRG is interested in resuming this type of research.

5. Transmutation fuel irradiations

Several irradiation tests were conducted within the European Experimental Feasibility of Targets for Transmutation (EFTTRA) network which advises EURATOM in matters related to transmutation fuel/targets in fast reactors or Accelerator Driven Systems. In the frame of the EURATOM Sixth Framework Program (FP6), the EUROTRANS project is currently preparing the HELIOS irradiation. This experiment will investigate the in-pile behavior of various uranium-free fuels and targets to determine the role of microstructure and temperature on gas release and fuel swelling up to high burn-ups.

Americium is one of the radioactive elements contributing to a large part to the radiotoxicity of spent fuels. Transmutation of long-lived nuclides like ^{241}Am by irradiation in nuclear reactors is an option for the reduction of mass and radiotoxicity of nuclear waste.

The transmutation scheme of ^{241}Am is shown in Fig. 10, from which it can be deduced that:

- Americium transmutation requires high neutron fluence for transmutation of all actinides present.
- The main fissile nucleus in the transmutation chain of ^{241}Am is ^{239}Pu.
- Three neutrons are required to induce fission of ^{239}Pu formed from ^{241}Am.
- ^{242}Cm emits a helium nucleus during decay.

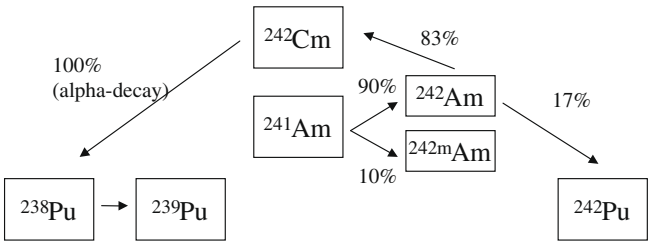

Fig. 10. Transmutation chain of ^{241}Am.

The analysis of previous irradiation experiments (e.g., EFTTRA T4 [8–10] and EFTTRA T4bis [11]) which were carried out at the HFR Petten with targets of $MgAl_2O_4$ + 11 wt% ^{241}Am showed that the release/trapping of helium is the key issue for target design. In fact, in those experiments significant volume swelling was observed which was attributed to the production of helium occurring during ^{241}Am transmutation. New fuel should experimentally demonstrate that helium releases from the target readily during irradiation thus reducing fuel swelling at the expense of pin pressurization.

For this purpose, a new irradiation experiment dubbed HELIOS was proposed. The main objective of HELIOS is to observe the in pile behavior of U-free fuels and targets such as ((Pu), Am, Y, Zr)O_2 or CerCer $Am_2Zr_2O_7$ + MgO or CerMet (Pu, Am)O_2 + Mo or (Zr, Y, Am)O_2 + Mo to reveal the role of microstructure and temperature on gas release and fuel swelling.

The fuel/target types to be irradiated in HELIOS follow two different approaches to enhance helium release:

(1) Provide helium release paths to the gas plenum. Pin 1 contains a composite target with MgO matrix featuring a network of open porosity.
(2) Increase target temperature to promote helium release from the matrix. Americium or americium/plutonium zirconia-based solid solutions (pins 2 and 3) along with CerMet targets (pins 4 and 5) will show the effect of temperature. The role of plutonium in association with americium is to increase target temperature at the beginning of irradiation.

The HELIOS irradiation will contain five experimental fuel/target pins with annular pellets (to accommodate a thermocouple) and solid pellets (cf. Fig. 11).

Pin 1: This pin contains MgO + $Am_2Zr_2O_7$ CerCer composite pellets to test the behavior of a pyrochlore oxide structure fabricated by CEA. The composition of this target is close to the EFTTRA T4 and T4bis targets. The spinel matrix is replaced by magnesia (MgO) because of its high resistance to irradiation damage. The MgO matrix with tailored porosity will provide helium release paths and accommodate swelling. The maximum temperature of targets was set to about 800 °C. The use of pyrochlore oxide ($Am_2Zr_2O_7$) as a support for minor actinides results from conclusions of earlier studies [12] indicating pyrochlore structures to be quite stable under irradiation. The diameter of $Am_2Zr_2O_7$ particles is in the range of 5–50 μm. Smaller sizes would give rise to chemical interaction between MgO and pyrochlore particles, whereas larger sizes would favor cracks in the MgO matrix during fabrication.

Pin 2: This is an instrumented pin with a central thermocouple containing (Am, Zr, Y)O_2 solid solution pellets (homogeneous fuel) fabricated by JRC-ITU. The cubic (Am, Zr, Y)O_2 phase is known to show very high stability under irradiation and quite low thermal conductivity. Yttria-cubic stabilized zirconia (Y-CSZ) is a refractory ceramic with a high melting point, enabling higher operating temperature (around 2500 K). Just like yttrium, the americium insert in the Y-CSZ crystal lattice can play the role of a stabilizing agent for

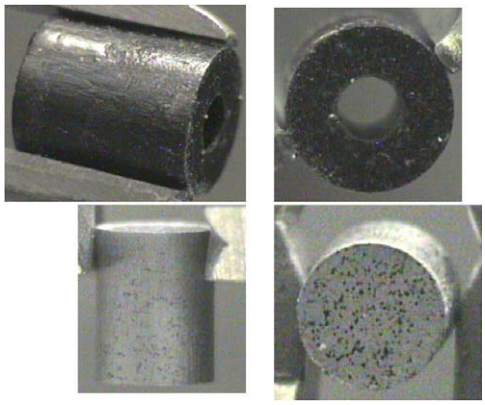

Fig. 11. Annular and solid fuel pellets to be irradiated in HELIOS (outer diameter: 5.45 mm).

the cubic phase. This target is to be directly compared with pin 3. The only difference between the two targets is the presence of plutonium in pin 3 to produce higher temperature.

Pin 3: This is again an instrumented pin with central thermocouple containing (Pu, Am, Zr, Y)O_2 solid solution pellets (homogeneous fuel) fabricated by JRC-ITU. This target has the same composition as pin 2 with addition of plutonium in the solid solution, i.e. (Am, Pu, Zr, Y)O_2. Plutonium is added to increase the initial power of the target, such that high temperature can be obtained at the beginning of irradiation. With this design, the formation and accumulation of helium bubbles can be avoided at an early stage, which is expected to reduce swelling.

Pin 4: This pin contains (Am, Zr, Y)O_2 + Mo CerMet composite pellets fabricated by JRC-ITU. The use of a molybdenum matrix strongly increases thermal conductivity of the target so that compared to the CerCer concept (i.e., pin 1), the operating temperature is expected to be quite low. Consequently, the target can hold a larger amount of actinides without dramatic temperature increase. Therefore, this target will be a 'cold' target with no extra open porosity and will serve as a comparison to the ceramic inert matrices. The good fission gas retention of the metal matrix must then be weighed against the observed degree of swelling.

Pin 5: This pin contains (Pu, Am)O_2 + Mo CerMet composite pellets fabricated by JRC-ITU. This target has a similar composition to pin 4 with addition of plutonium in the solid solution and elimination of zirconium and yttrium. Use of a Mo-based CERMET with (Pu, Am)O_2 as actinide compound maintains the advantage of good thermal conductivity (i.e., low irradiation temperature). However, in contrast to the zirconia-based compounds, this fuel can be reprocessed and allows multi-recycling thereby achieving a higher burn-up of radiotoxic isotopes (see Table 1).

Due to the asymmetric thermal–mechanical load a standard design configuration had to be modified for a scheduled start-up in the end of 2008.

Further irradiation tests (AMEX and CURIOS) are in the conceptual design phase. AMEX will demonstrate the feasibility of full transmutation (95–99%) of americium whereas CURIOS will focus on the behavior of curium targets. This type of fuel/target is expected to be used either in Fast Reactors or Accelerator Driven Systems for incineration of minor actinides from Light Water Reactors.

6. Thorium fuel

Thorium (^{232}Th) is a fertile material which converts into fissile ^{233}U during irradiation. The subsequent fission of ^{233}U produces less long-lived transuranium elements (Np, Pu, Am, Cm) than the conventional ^{235}U fuel cycle and is thus an interesting option for waste reduction. Moreover, thorium ore is more abundant in the earth's crust than uranium, and more evenly distributed. The inconvenience of the thorium fuel cycle is that reprocessing is mandatory and that certain reprocessing steps require remote control and extra shielding. Consequently, the thorium fuel cycle is of interest for countries with large thorium ore reserves, but of lower priority for countries with large stocks of depleted uranium (e.g., France) which tend to favor fast breeder technology based on the U–Pu fuel cycle.

Since uranium prices are soaring, thorium is gaining momentum internationally. For instance, India is actively exploring the thorium fuel cycle for power generation.

In the THORIUM experiment (2001–2003 [13]), four fuel pins were irradiated to high burn-up (>50 GWd/tHM) so as to supply key data for use in Pressurized Water Reactors, fast reactors and Accelerator Driven Systems. The irradiation lasted for 2.5 years (699 efpd) with the objective of comparing the behavior of thorium-based fuels, on which relatively little information is available, to uranium-based fuels [14]. The maximum allowable temperatures were 450 °C at the surface of the cladding and 1200 °C at the central thermocouple, together with a maximum fission power of 280 W/cm or 1200 W/cm^3.

The pellets had a diameter of 5.48 mm and consisted of ThO_2, UO_2, $(U_{0.9}Pu_{0.1})O_2$ and $(Th_{0.89}Pu_{0.11})O_2$. These pellets (fuel stack length 100 mm) were instrumented with thermocouples and were sealed into 15Cr–15Ni–Ti stainless steel cladding with an outer diameter of 6.55 mm and a wall thickness of 0.45 mm. The resulting pins were initially pressurized with 0.1 MPa helium with the plenum length designed to keep the maximum gas pressure below 2.5 MPa. The pins could be remotely centered in the maximum axial neutron flux to minimize power gradients.

The four pins were inserted into a thimble of a standard TRIO-131 rig, providing space for three independent experiments in a single in-core position with a useful diameter and length per thimble of 31.5 and 600 mm, respectively. Fig. 12 shows a cross section of the in-pile part of the rig with the THORIUM sample holder containing the four fuel pins on the lower left side.

Instrumentation consisted of 24 thermocouples, two neutron fluence detectors and five gamma scan wires. Thermal bonding was achieved by a static sodium bath. Failure of numerous thermocouples during irradiation led to somewhat premature termination of the experiment for safety reasons. Nevertheless, the burn-up

Table 1
Fuel and targets for the HELIOS irradiation experiment in HFR.

Pin	Target	Microstructure	100% TD density (g/cm^3)		% Pellet TD measured	As-fabricated density (g/cm^3)		Instr.
			^{241}Am	Pu-total		^{241}Am	Pu-total	
1	$Am_2Zr_2O_7$ + MgO	5–50 μm	0.77		91.5	0.66		
2	(Am, Zr, Y)O_2	Solid solution	0.76		92.6	0.70		TC
3	(Am, Pu, Zr, Y)O_2	Solid solution	0.82	0.46	89.7	0.74	0.39	TC
4	(Zr, Am, Y)O_2 + Mo	65–125 μm 71.3 vol.% Mo	0.74		94.1	0.70		
5	(Pu, Am)O_2 + Mo	>150 μm 84.2 vol.% Mo	0.31	1.29	96.0	0.30	1.20	

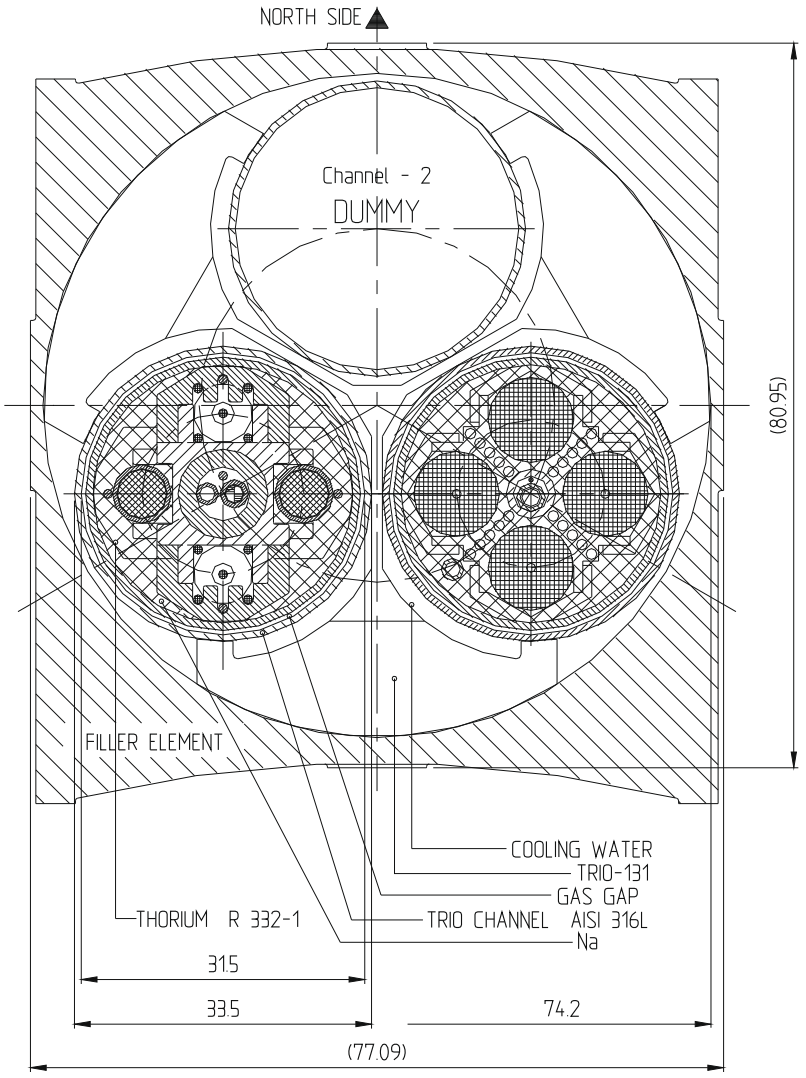

Fig. 12. Cross section of THORIUM in TRIO-131 rig (capsule outer diameter: 33.5 mm).

objective was achieved with approximately 62 GWd/tHM. Post-irradiation examination on these pellets is currently being completed.

7. Conclusion

Existing expertise and recent experimental activities at the HFR Petten support international R&D on next generation reactor concepts, fuel and fuel cycles. Over the years, the HFR has successfully maintained its reputation as a versatile infrastructure tool for nuclear technology. This includes fuel and material tests for fission and fusion, silicon doping, medical isotope production, neutron capture therapy, various neutron methods, etc. JRC and NRG installations share the Petten site and have created significant synergies for the benefit of commercial customers and for international R&D programs.

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