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Operation and Utilisation of the High Flux Reactor

Annual Report 2012

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Operation and Utilisation of the High Flux Reactor
Annual Report 2012

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1 HFR Operation

1.1 Operating Schedule

In 2012, the regular cycle pattern consisted of a scheduled number of 296 operation days and one maintenance period of 31 days in March. The In-Service Inspection of the north and south reducer, welds of the reactor vessel and the annual leak test of the reactor containment were performed during this period. In reality, the HFR has been in operation 253 days (Table 1, Figure 1). This corresponds to an actual availability of 85.26 % with reference to the original schedule. Nominal power has been 45 MW with a total energy production in 2012 of approximately 11 313 MWd, corresponding to a fuel consumption of about 14.12 kg U-235. The planned cycles 2012-10 and 2012-11 were cancelled. One of the reasons for this was the detection of tritium in the groundwater around the reactor building traced back to an underground leak in a water pipeline. The other reason was the detection of a leak path between the primary cooling water system and the bottom plug cooling system (part of the pool cooling system). Both issues were investigated and will be repaired.

Generated Energy [MWd]	Planned [h]	Low Power [h:min]	Nominal Power [h:min]	Other Use [h:min]	Total [h:min]	Planned [h:min]	Un-planned [h:min]	Stack Release (of Ar-41) [10^{11} Bq]
11313.55	7144	21:43	6028:14	41:00	6090:57	1476:41	1216:22	38.12
Percentage of total time in 2012 (8784 h):		0.25	68.63	0.47	69.34	16.81	13.85	
Percentage of planned operating time (7144 h):		0.30	84.38	0.57	85.26			

Table 1: Summary of HFR Operation in 2012

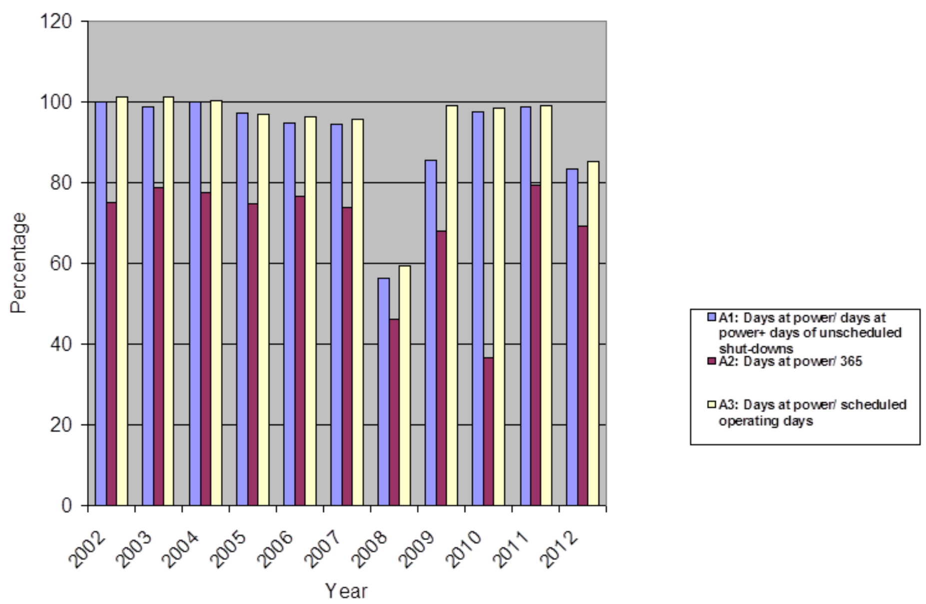


Figure 1: HFR availability since 2002

During the reporting period, the annual 30 MW reactor training for the operators was carried out in October. After the scheduled end of 45 MW operation of each cycle, these cycles were directly followed by activities performed in the framework of the regular HFR operator training.

Figure 2 shows the total discharged activity of tritium and noble gases in 2012 compared to values since 1999.

The license limit is 100 RE/year. The total discharged activity in 2012 was approximately 11 RE.

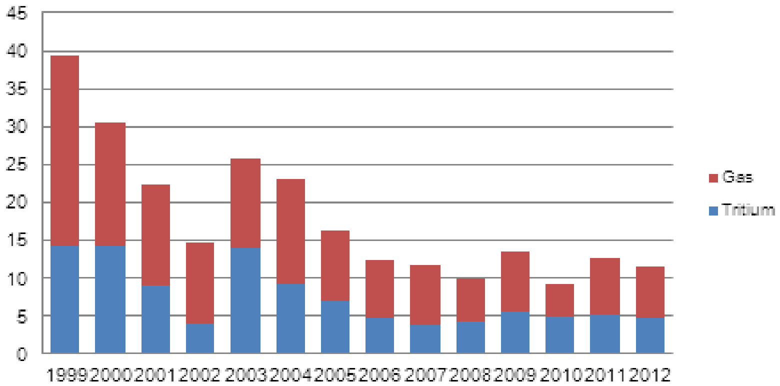


Figure 2: Radioactivity (noble gas and tritium) discharged since 1999 (licence limit 100 RE/year)

1.2 Maintenance and Engineering Activities

In 2012, the maintenance activities consisted of the preventive, corrective and breakdown maintenance of all Systems, Structures and Components (SSC) of the HFR as described in the annual and long-term maintenance plans. These activities are executed with the objective to enable the safe and reliable operation of the HFR and to prevent inadvertent scrams caused by insufficient maintenance.

The main activities during the maintenance period in March comprised:

- The periodic leak test of the containment building as one of the licence requirements (0.5 bar overpressure during 48 hours).
- The In-Service Inspection of the north and south reducers and the welds of the reactor vessel.
- Regular repair maintenance of the concrete pipeline for secondary cooling water between the North-Holland Canal to the HFR secondary pump building.
- Completion of the remote monitoring system which is used to monitor important reactor parameters during emergency cases.
- Completion of the alternative shutdown system which can be used in case the normal shutdown system is not functioning.

These activities were successfully completed.

2 The HFR as a Tool for Research on Reactors, Materials and Fuel Cycles

2.1 Towards a fuel cycle with less nuclear waste: The FAIRFUELS Project

In the frame of the EURATOM 7th Framework Programme (FP7), the 4 year project FAIRFUELS (Fabrication, Irradiation and Reprocessing of FUELS and targets for transmutation, http://www.fp7-fairfuels.eu/) aims at reducing the volume and hazard of high level radioactive waste by incinerating the most long-lived isotopes to shorter-lived isotopes. In this way, the nuclear fuel cycle can be closed in a more sustainable manner. The FAIRFUELS consortium consists of ten European research institutes, universities and industry. The project started in 2009 and is coordinated by NRG. NRG and JRC-IET work closely together on several irradiations tests scheduled in FAIRFUELS.

2.1.1 MARIOS

Objective:

MARIOS, as part of the FAIRFUELS project, is an irradiation test dealing with heterogeneous recycling of Minor Actinides (MA) in sodium-cooled fast reactors (i.e. the MA-bearing blanket concept), cf. Figure 3. MA, such as americium and curium, are long-lived radioactive isotopes in high level nuclear waste, which are currently not recycled. The aim of MARIOS is to investigate more closely the behaviour of MA targets in a uranium oxide matrix and to compare dense fuel to fuel with a tailored porosity (cf. Figure 4). In these targets, large amounts of helium are produced, which cause swelling and significant damage to the material under irradiation. It is the first time that americium (^{241}Am) is included in a (natural) uranium oxide matrix $Am_{0.15}U_{0.85}O_{1.94}$ to conduct an experiment in order to assess helium production and swelling.

Achievements:

The MARIOS irradiation started on 19 March 2011 and successfully finished on 2 May 2012 after 11 reactor cycles (~304 full power days) in position G7 of the HFR core. The irradiation was completed and disassembled in the NRG hot cells in Petten and is ready for Post Irradiation Examination (PIE) as part of another European project, PELGRIMM (www.pelgrimm.eu). During this phase, the released He fraction will be determined and various techniques will be employed to assess damage to the fuel and the redistribution of isotopes within the pellets.

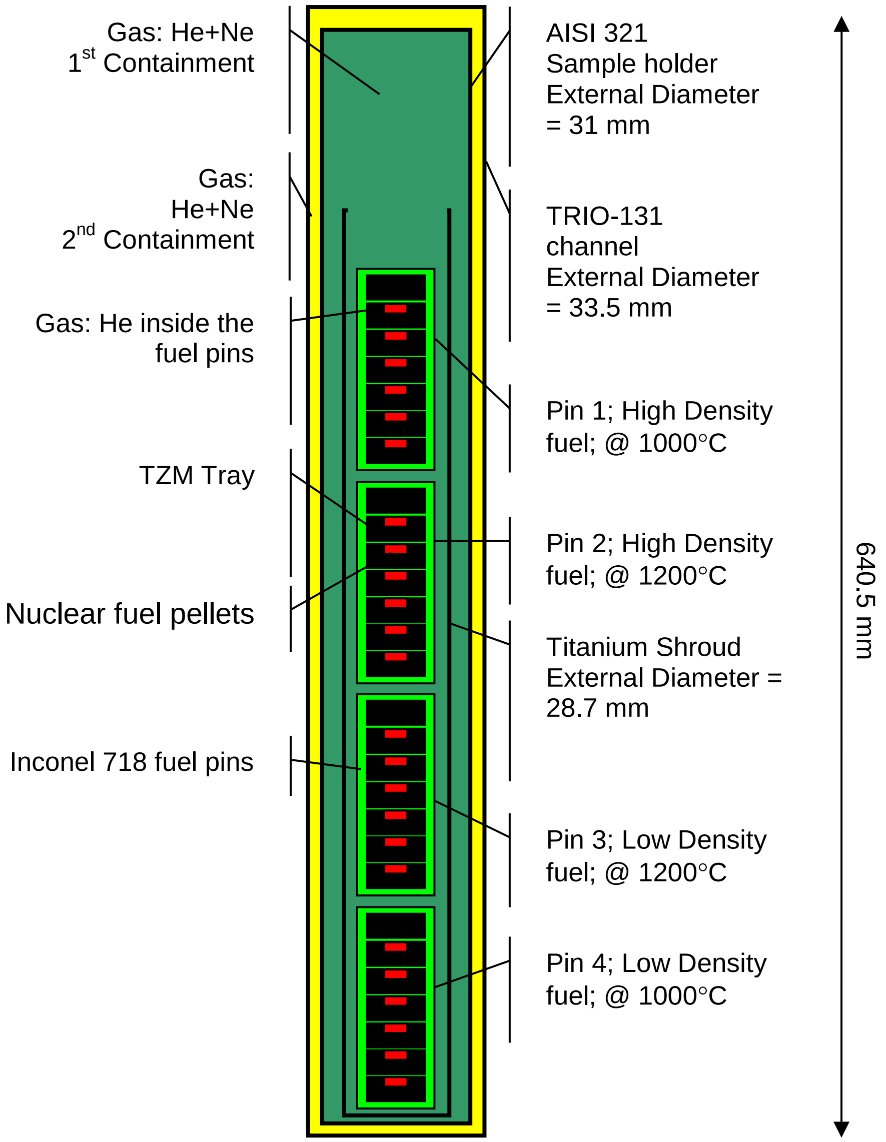


Figure 3: Schematic view of the MARIOS sample holder

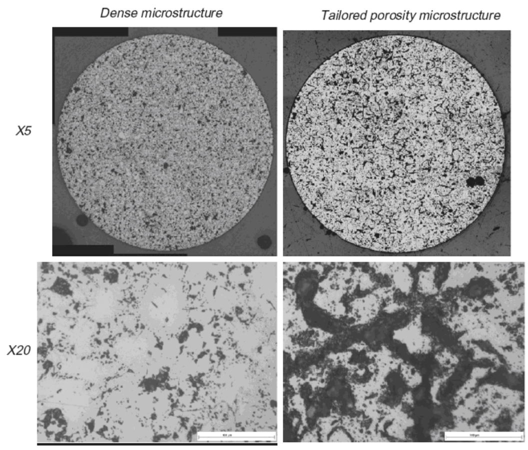


Figure 4: Pre-irradiation microstructure of the MARIOS pellets

2.1.2 SPHERE

Objective:

Another test planned as part of the FP7 FAIRFUELS project is the irradiation test SPHERE which was designed to compare conventional pellet-type fuels with so-called Sphere-Pac fuels under irradiation (cf. Figure 5). The latter have the advantage of an easier, dust-free fabrication process. When dealing especially with highly radioactive minor actinides, dust-free fabrication processes are essential to reduce the risk of contamination. For this purpose, americium containing fuel, both pellet-type and Sphere-Pac-type, will be fabricated by JRC-ITU in Germany. These fuels will be irradiated in the HFR in a dedicated test-facility. It is the first irradiation test of this kind, as MA bearing Sphere-Pac fuel has never been irradiated before. The SPHERE irradiation is planned to start in 2013 for approximately 300 full power days.

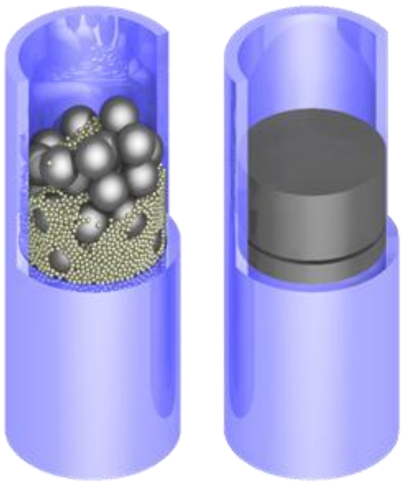


Figure 5: Sphere-Pac versus pellet concept

Achievements:

During 2012, the design of SPHERE was finalised, the fuel was manufactured by JRC-ITU and delivered to NRG in summer 2012. The manufacturing, assembly and commissioning of the irradiation experiment was completed so that SPHERE is ready for irradiation in position G7 once the HFR will resume operation.

2.1.3 HELIOS

Objective:

The HELIOS irradiation was performed as part of the terminated FP6 EUROTRANS Integrated Project on Partitioning and Transmutation (2005-2010) and dealt with irradiation of U-free inert matrix fuels containing americium with the objective to incinerate this MA. The main objective of the HELIOS irradiation was to assess the in-pile behaviour of such fuel targets such as CerCer $(Pu,Am,Zr)O_2$ and $Am_2Zr_2O_7$+MgO (ceramic matrices) or CerMet $(Pu,Am)O_2$+Mo (metallic matrix), in order to gain knowledge about the role of microstructure and temperature on helium gas release and on fuel swelling. During the irradiation of such fuel, a significant amount of helium is produced due to the transmutation of americium. The understanding of the gas release mechanisms is vital to maximize the transmutation yield and to achieve optimum performance of these fuels. Two different approaches were followed to reach early helium release and thus to keep fuel damage low:

1. Provide release paths for helium to plenum gas by creating open porosity in the fuel. Therefore, in the HELIOS test plan a composite target with an MgO matrix containing a network of open porosity was included.
2. Increase target temperature in order to promote the release of helium from the matrix. Americium or americium/plutonium zirconia based solid solutions along with CerMet targets were included in the test plan to investigate the effect of temperature. The role of the added plutonium in association with americium is to increase the temperature of the target already at the beginning of irradiation.

PIE of irradiated HELIOS fuel was planned as part of the FP7 FAIRFUELS project.

Achievements:

In 2012, the PIE were finalized in the NRG hot cells. All 5 pins were punctured to analyse the gas pressure (to determine the helium release fraction) and the isotopic distribution in the fuel. Destructive PIE was performed in collaboration with JRC-ITU and CEA. The pin with MgO matrix was to be examined by CEA but transport issues prohibited this. CEA therefore visited NRG to perform the examinations in collaboration. The pin showed very stable behaviour where indeed the increased porosity stimulated the release of fission gas, around 43%, most of which was helium. The swelling of the pellets was very acceptable with a volumetric swelling of less than 3% for a burn up of 5.11%.

For the pins containing molybdenum matrix fuel, optical microscopy and SEM examinations were performed at NRG. For these pins, a clear temperature effect was observed. Where the pin irradiated at roughly 500°C showed very stable behaviour, almost no swelling and limited gas release, the pin at high temperature (up to 1200°C) showed the opposite. Although favourably high helium gas release was measured, this was insufficient to avoid significant fuel swelling so that pellet-cladding interaction occurred.

The zirconia based pins were transported to JRC-ITU where further optical microscopy and SEM work will be done. The transport was performed in March 2012.

2.2 Towards a safer and more efficient closed fuel cycle: The PELGRIMM Project

In the frame of the EURATOM 7th Framework Programme (FP7), the 4 year project PELGRIMM (PELlets versus GRanulates: Irradiation, Manufacturing & Modelling) has the objective to investigate the implementation of MA-bearing fuels as granulates instead of pellets.

The granulate fuel would lead to a significant simplification of the fabrication method thanks to the elimination of some process steps such as milling, pressing and grinding, that involve the handling of fuel powders (and dust), and the remaining steps may be carried out relatively easily in shielded environments. Moreover, the fuel performance under irradiation may be improved as well, thanks to a better accommodation of solid swelling (compared to pellets) through the re-arrangement of the free inter-particulate areas and possibly enhanced release of helium and fission gases thus reducing fuel swelling and damage.

The PELGRIMM consortium consists of twelve European research institutes, universities and industry. The project started in 2012 and is coordinated by CEA. Both NRG and JRC-IET work closely together on the HFR irradiations that are scheduled in PELGRIMM.

2.2.1 MARINE

Objective:

Within the FP7 PELGRIMM project, the irradiation test MARINE is planned. MARINE was designed to compare conventional pellet-type fuels with the so-called Sphere-Pac fuels described above (cf. Figure 6). The goal of the MARINE irradiation is to determine helium release behaviour and fuel swelling in $^{241}Am_{0.15}U_{0.85}O_{2-x}$, which is representative of a MA-bearing blanket material to be used for transmutation in sodium-cooled fast reactors.

Americium containing fuel samples will be fabricated at JRC-ITU in Germany. This fuel will be irradiated in the HFR using a dedicated test facility. The irradiation will be equipped with internal pressure sensors monitoring online the release of helium, which is characteristic of this kind of americium-containing fuel. The MARINE is planned to start in early 2014 for approximately 300 full power days.

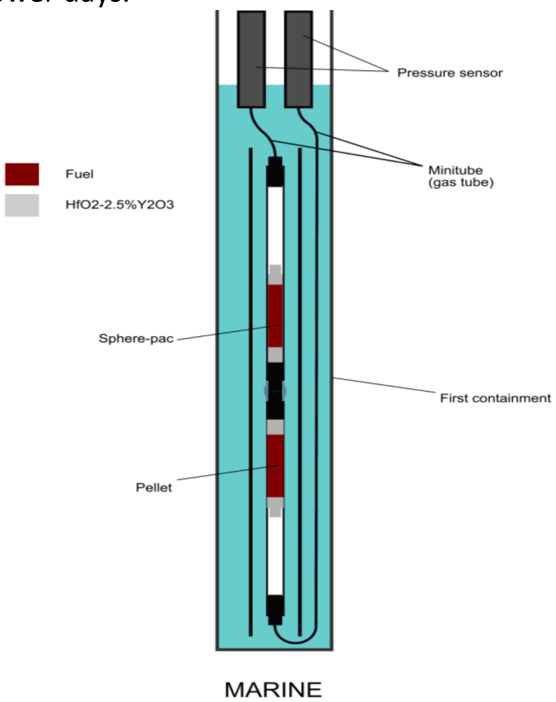


Figure 6: Schematic view of the MARINE irradiation experiment

Achievements:

During 2012, the preliminary design of MARINE was finalized with the development of a new system to connect the pressure transducer (fabricated in Halden, Norway) with the fuel pin (fabricated at JRC-ITU), as well as a first round of design calculations aimed at understanding how well the conditions in the SFR blankets (power, temperature, helium production) can be reproduced in MARINE.

2.3 Fuel and Graphite Qualification for High Temperature Reactors

High Temperature Reactors (HTR) are being investigated in a number of countries as a safe and efficient source of energy, in particular for cogeneration of industrial process heat and electricity. Related new demonstration projects are either existing or envisaged in several countries (e.g. Japan, China, US, South Korea) and are subject to current R&D in Europe. The HFR is used in particular for the qualification of fuel and graphite which are decisive elements for the benign safety performance of this type of reactor.

2.3.1 HFR-INET

Based on previous experience in the 1980s related to the German HTR programme in which the HFR had played already a crucial role, a licensing strategy was developed encompassing: (1) fuel irradiation at relevant temperatures to a specified target burn-up and (2) successive heating tests of the irradiated fuel under simulated accident conditions. For the fuel to be considered acceptable in both types of tests, the measured fractional release of fission gas must remain below the licensing thresholds.

The Institute of Nuclear and New Energy Technology (INET) of the Tsinghua University in Beijing, China is currently constructing the Chinese Modular High Temperature Gas-cooled Reactor Demonstration Plant (HTR-PM). The fuel for HTR-PM is developed and manufactured by INET. INET requires qualification of their fuel to support licensing of HTR-PM.

The HFR-INET irradiation is the first step of this HTR fuel qualification for HTR-PM and is being performed by NRG. Five spherical HTR fuel elements (“pebbles”) are irradiated under controlled conditions in the HFR, at almost constant central pebble temperature, while fission gas release is measured continuously online using the sweep loop facility. This sweep loop facility was in the past developed and used by JRC-IET and is currently employed by NRG. This facility enables temperature control of the fuel elements and monitoring of fission gas release during irradiation by gamma spectrometry. Fission gas release is an important measure for fuel performance and quality under operational conditions, and forms an essential part of the fuel qualification. For the qualification irradiation, a dedicated irradiation test facility was designed and manufactured. The irradiation started in September 2012 in a high flux in-core position of the HFR and will continue until the required burn-up is achieved. After irradiation, non-destructive PIE will be performed in the NRG hot cell laboratories.

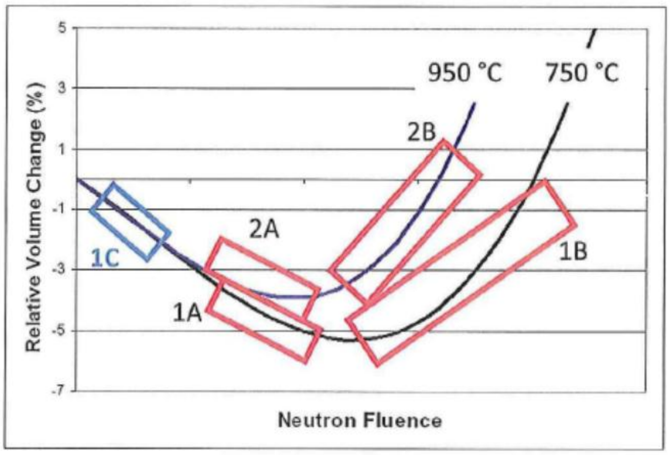
For the second step of the fuel qualification, the five HTR pebbles will be transported to JRC-ITU where they will be subjected to heating tests in the so-called KÜFA facility. The heating tests shall verify the integrity and proper performance of irradiated HTR fuel even under simulated accidental conditions.

2.3.2 INNOGRAPH-1C

Graphite is used as moderator and reflector material in an HTR and is known to first shrink and then swell under irradiation. This behaviour depends on temperature, neutron dose and graphite grades. Its understanding is required to enable proper design of such reactors and to put the graphite manufacturing industry in a position to produce suitable graphite grades with stable properties over longer periods of time.

The INNOGRAPH-1C irradiation is performed as part of the FP7 ARCHER project (Advanced High-Temperature Reactors for Cogeneration of Heat and Electricity R&D, www.archer-project.eu). Following earlier irradiation tests (cf. Figure 7), it will complete the data set for different graphite grades at different temperatures, under a range of neutron doses. The experiment is a technical building block for nuclear cogeneration using HTRs as an alternative to fossil fuels.

The experiment was successfully commissioned for irradiation in 2012. Irradiation will continue in 2013 after which a PIE programme will take place on the graphite samples.



INNOGRAPH tests:

1A: 750°C, low/medium dose (FP5)

1B: 750°C, high dose (RAPHAEL)

2A: 950°C, low/medium dose (RAPHAEL)

2B: 950°C, high dose (RAPHAEL)

1C: 750°C, low dose (ARCHER)

Figure 7: INNOGRAPH-1C with focus on early part of the shrink-swell curve

2.4 Materials Irradiations

2.4.1 BLACKSTONE

The United Kingdom operates a fleet of Advanced Gas Cooled Reactors (AGR) operated by EdF Energy. Graphite degradation is considered to be one of the key issues that determine the remaining service life of the AGR. Graphite data at high irradiation dose and weight loss is required, to allow prediction and assessment of the behaviour of AGR graphite cores beyond their currently estimated lifetimes thus ensuring continued safe operation and lifetime extension. The BLACKSTONE irradiations use samples trepanned from AGR core graphite and subject them to accelerated degradation in the HFR. The tests are designed to enable the future condition of the AGR graphite to be predicted with confidence.

After the irradiations of BLACKSTONE Phase I, which finished in 2010, the final measurements were performed on these samples in 2012 and the results were successfully used in the safety case by EdF Energy. In the meantime, two experiments were started in Phase II. This time both of them are connected to the Gas Handling Facility to monitor the amount of graphite oxidized online during irradiation. Of these experiments, the first has reached the target of 12 HFR cycles in October 2012. The dismantling was performed successfully, the PIE campaign has started and is expected to be finalized in the end of 2013. The 4th irradiation is ongoing and expected to reach its target after 16 irradiation cycles.

2.4.2 LYRA-10

The LYRA irradiation rig is used in the framework of the European AMES (Ageing Materials and Evaluation Studies) Network activities with the main goal of understanding the irradiation behaviour of reactor pressure vessel (RPV) steels, thermal annealing efficiency and sensitivity to re-irradiation damage. The LYRA-10 experiment placed in the Pool Side Facility (PSF) of the HFR consists in the irradiation of different specimen types representative of reactor pressure vessel materials, namely model steels, realistic welds and high-nickel welds (cf. Figure 8). The model steels comprise of 12 batches with the basic, typical composition of WWER-1000 and PWR reactor pressure vessel materials studied by JRC-IET with the scope of understanding the role and influence of Ni, Si, Cr and Mn as alloying elements and certain impurities such as C and V on the mechanical properties of steels. The realistic welds are created at eight different heats, specially manufactured on the bases of typical WWER-1000 weld composition with variation of certain elements, such as Ni, Si, Cr and Mn. They are of importance to investigate the role and synergisms of alloying elements in the radiation-induced degradation of RPV welds. The LYRA-10 irradiation campaign started in May 2007 and up to now underwent six HFR cycles at an average temperature of 283°C with an accumulated fast neutron fluence in the samples of ~3.4×10^{23} n m^{-2} (E>1 MeV). Originally planned to be irradiated for 7 more cycles to achieve a fast fluence of ~6×10^{23} n m^{-2} (E>1 MeV), it was decided during the LYRA-10 outage that at least 10 more HFR cycles are required to determine a hypothetical ”late-blooming” effect that may occur in the irradiated materials. "Late blooming" refers to a possibly increased effect at higher irradiation doses on material properties, e.g. a stronger decrease of ductile-to-brittle transition temperature in materials which would not follow currently used correlations.

In order to proceed with the resumption of the LYRA-10 experiment, some revamping actions were performed in 2012 to make the rig fit for irradiation. The experiment should have started at the end of 2012 but due to the HFR stop, it has been delayed into 2013.

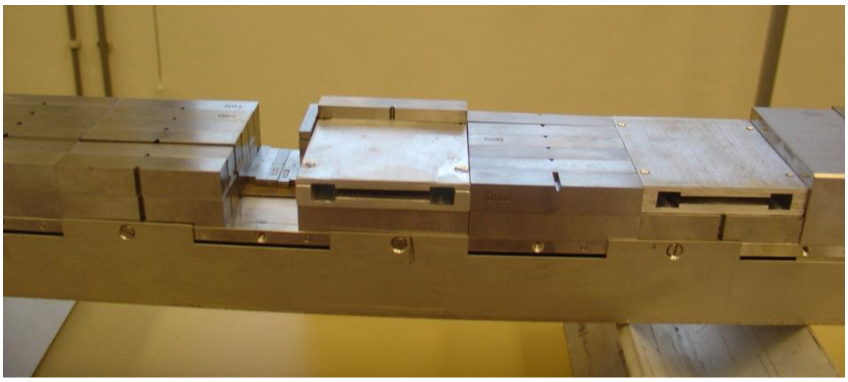


Figure 8: LYRA-10 during assembly in an irradiation rig

2.5 Perspectives for future fuel and material irradiation tests

In the following, an outline is given of several future R&D items requiring irradiation testing. They are in line with pre-existing knowledge at NRG and JRC and were identified as key safety aspects of advanced reactors.

Fuel-clad interaction:

The integrity of fuel rods in normal and accidental conditions is essential for nuclear safety. The fuel itself constitutes the first barrier against radioactivity release and the cladding represents the second barrier. Clad failure may thus set the limit for the technologically achievable burn-up. For meaningful safety assessments, the fuel and its clad need to be considered as one component.

Fuel-clad interaction (FCI) can be classified into mechanical (fuel swelling leads to contact with the clad) and chemical (mainly due to chemical reaction at high temperature between fission products and the clad). The interaction depends on the irradiated properties of both the fuel and the clad. Moreover, the critical loading is often complex. Finally, chemical attack and mechanical loading may combine to promote stress corrosion cracking (SCC) of the cladding. Fuel-clad interaction has been and is extensively studied for Gen II and Gen III reactors. Under the acronyms FCI or PCI (Fuel/Pellet-Clad Interaction) extensive scientific literature can be found describing both chemical and mechanical interaction, in particular for classical UO_2 fuel with zircaloy clad.

FCI has been much less studied for Gen IV fast reactors. The clad materials (15-15 Ti steel, possibly P91 and in the future ODS steel) and fuel (mixed oxides, carbides and nitrides) are different from the ones used in Gen II and III reactors. In the short term, the driver fuel for fast reactors will be uranium-plutonium mixed oxide (MOX), but an important longer term option is the burning of MA in the core (homogeneous concept) and/or in a blanket surrounding the core (heterogeneous concept). Several material options and combinations are available for this. Due to the high temperatures and temperature gradients experienced by such fast reactor fuel, chemical interaction tends to become more and mechanical interaction less important compared to LWR.

Therefore, FCI for fast reactors is considered an incompletely-solved issue and of significant interest, both from a JRC and NRG point of view, for future testing in the HFR.

Irradiation-induced creep in MOX fuel:

Irradiation-induced creep is the deformation of a material under the combination of an external load and irradiation in a reactor. In nuclear fuel it is considered to be driven by nuclear fission reactions. Related research will focus on low-temperature creep in order to remove the impact of thermal creep. It is intended to make a comparison between the creep behaviours of heterogeneous and (chromia-doped) homogeneous MOX fuel. A section of fuel inside a cladding segment might be added to additionally study FCI.

Transient experiments on fast reactor fuel:

Transient experiments on fast reactor fuel are considered particularly relevant for safe reactor operation. Therefore, it is intended to perform in-pile studies of fuel melting behaviour through ramp tests in the HFR Pool Side Facility where the heat generation in irradiation rigs can be modified rapidly by sliding the rig to and away from the HFR core. These tests might for instance focus on fuel relocation in fast reactor fuel which in some scenarios is needed to warrant sub-criticality. This requires molten fuel to flow downward through the central hole of the annular pellets typically used in fast reactors. Various types of fuel should be investigated, including MA-bearing fuel or targets. A base irradiation of the fuel might be needed prior to the transient.

Irradiation resistance of coatings used in coated particle fuel:

The safety performance of nuclear reactors using coated particle fuel (in particular High Temperature Reactors) depends to a high degree on the integrity of the used coatings under irradiation conditions including internal pressurisation of the particles with fission gases and helium. To improve existing fuel performance codes required for licensing, a separate effect test is considered, where dummy kernels without fuel are coated or infiltrated with boron carbide before being coated with a standard TRISO coating made of layers of pyrocarbon and silicon carbide. Under neutron irradiation, the boron carbide produces helium thus internally pressurising the coated particles which simultaneously undergo irradiation damage. Using suitable PIE methods, the properties of coatings having received different doses at different temperatures can be determined thus improving the fuel performance codes.

2.6 Irradiations for Fusion Technology

2.6.1 ITER PRIMUS

Objectives:

In 2005, an irradiation experiment was defined with the European Fusion Development Agency (EFDA) to test thermal fatigue during irradiation of normal heat flux modules for the ITER first wall. This had been foreseen to be performed in an HFR Pool Side Facility position for a duration of 22 cycles. The experiment had to be designed in a way that the stress conditions and temperatures would reflect ITER first wall conditions. Between 2005 and 2007, multiple iterations and adjustments have been made to achieve the ITER conditions. The iterations led to a final design in 2008, but could not be pursued due to HFR outage. In 2009, the required PSF position was not available. In 2010, the Reactor Safety Committee gave the feedback that the thermal cycling in the HFR was not possible due to the intrinsic design of the HFR Automatic Control Rod system. It has been discussed with F4E to perform a stagnant in-pile experiment instead to achieve 1 dpa in beryllium and to perform the thermal cycling afterwards in the JUDITH facility at Forschungszentrum Jülich, Germany.

To perform this irradiation, a new experiment was designed called PRIMUS. Here the first wall mock ups will be loaded in a REFA facility and will be irradiated for 5 cycles in position H2.

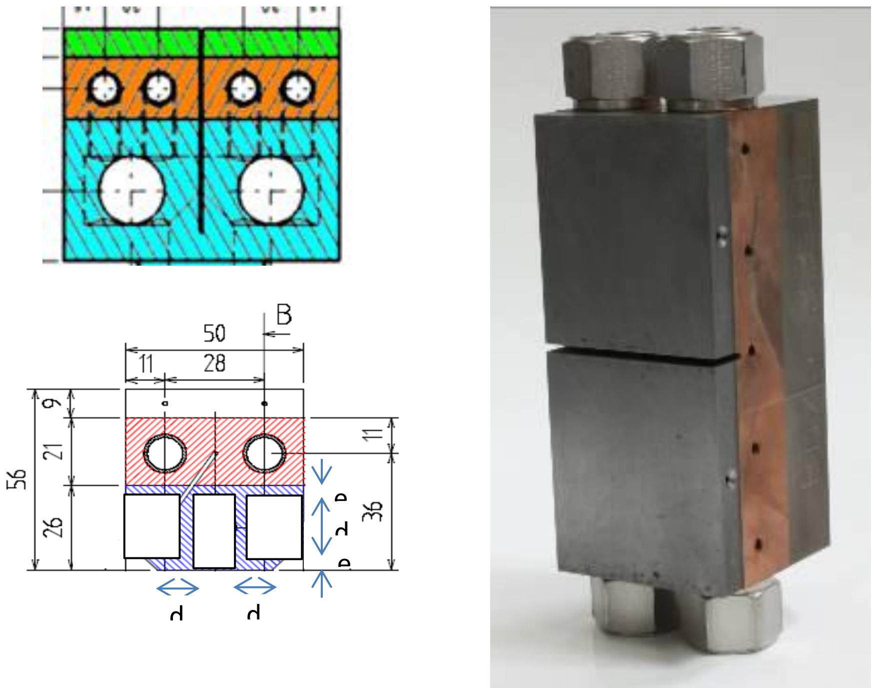


Figure 9: Left top: Normal heat flux module design, left bottom: cutting plan, right: uncut module

Achievements:

In 2012, a new conceptual design has been presented to F4E. Extensive activation calculations showed that the mock ups are too active to be handled in the JUDITH facility. Therefore, a proposal has been made to cut the steel back from the mock-ups, without jeopardising the stress conditions on the beryllium/CuCrZr interface.

This led to a proposal which reduced the activation to levels acceptable for JUDITH. The mock-ups were sent back to France were the cutting procedure has started. The mock-ups will be sent to FZJ for first screening and are expected at NRG in May 2013. The irradiation is scheduled to start in November 2013.

2.7 The HFR in support of standardisation in materials research

Network on Neutron Techniques Standardisation for Structural Integrity (NeT)

The European Network on Neutron Techniques Standardisation for Structural Integrity (NeT) fosters performance and safety improvements in European nuclear power plants. In spring 2012, NeT celebrated its 10th anniversary during the 21st Steering Committee Meeting held at the premises of TWI Ltd. in Cambridge, UK. The JRC organizes and manages NeT and contributes to the scientific work through neutron scattering for residual stress measurement using its beam tube facilities at the HFR. About 40 organisations from Europe and worldwide contribute actively to the work programme of NeT, with the meeting in Cambridge being the first to be attended by two representatives from the United States.

During the reporting period, NeT has developed new activities on residual stresses in welds of nickel-based alloys and aluminium specimens. These materials are relevant for new types of dissimilar metal welds in nuclear piping systems and to welds in research reactor installations.

Standardisation of the neutron diffraction method for residual stress measurement

The scientific and engineering community employ neutron diffraction as a technique for measuring residual stresses in materials and components. Work on the development of a standard for the method has been in progress for about 15 years. JRC has been involved from the beginning based on its two dedicated diffractometers at the HFR. The activity is now entering its final phase, whereby the current technical specification of the method is being upgraded to an international standard.

In 2012, negotiations were undertaken with ISO and several national neutron beam facilities in order to establish a Working Group that would be entrusted with the task of thoroughly reviewing the existing specification. It is expected that this Working Group will be established in 2013 and will deliver the final draft for ISO consideration in the second half of 2014.

3 Isotope Production

3.1 Production Performance

A normal operational year for isotope production in the HFR was achieved in 2012, up to mid-November, when the reactor was stopped. Consequently, only 9 full cycles of normal isotope production were achieved and 1.5 cycles of production were lost.

In the period until the HFR stopped, the value of isotopes and associated services supplied was higher than in the preceding year, once again demonstrating that the HFR plays an important role as the largest producer of medical isotopes in Europe and one of the largest producers in the world.

A number of interesting new product development ideas continued to progress, both in conventional application areas, as well as in some ground-breaking areas of medical technology.

Following a full year in 2011, when Neutron Transmutation Doped (NTD) silicon came back in to production, production during 2012 continued with the same pattern. NTD silicon products can be found in important applications such as high voltage power electronics, high speed trains and green technologies, e.g. wind, solar and hybrid cars.

In 2012, NRG continued to work closely with other players in the medical isotope supply network, as well as with the medical community, governments, the European Commission, AIPES, the OECD/NEA and the IAEA. These actions were particularly stepped up after the recent HFR outage when it was necessary for the players in the medical isotope supply chain to cooperate closely to ensure that supply shortages were avoided wherever possible.

The recent HFR outage in November 2012, confirmed further the need to support the coordinated efforts necessary to minimize the future risks to security of supply of critical medical isotopes. NRG and the HFR fully support the recommendations of the OECD/NEA High Level Group on the security of supply of medical isotopes and continue to work together with other international stakeholders on important issues such as full-cost recovery pricing, outage reserved capacity provision, future infrastructure investment and conversion to Low Enriched Uranium (LEU) targets for Mo-99 production.

3.2 Qualification of Low Enriched Uranium targets for Mo-99 Production

High Enriched Uranium (HEU) is generally considered to be the nuclear material most sensitive to unwanted proliferation. Therefore, a global effort is being undertaken to reduce, and eventually abandon, the use of HEU in the civil nuclear field.

Currently, targets irradiated for the production of molybdenum-99 (Mo-99), primarily used for nuclear medicine imaging, have contained HEU. In 2010, NRG, Covidien and CERCA started together a programme to fully convert the targets for Mo-99 production to Low Enriched Uranium (LEU), i.e. an enrichment level of less than 19.75% U-235. This conversion programme includes the qualification of the new LEU targets for usage in the High Flux Reactor. The LEU target specification was defined in close collaboration with NRG's partners: CERCA for target manufacturing and Covidien as the processor of the irradiated targets. The design of the target was chosen to maintain conventional manufacturing techniques and to be compatible with the irradiation conditions available in the HFR.

A dedicated irradiation test facility was designed and manufactured for the qualification of LEU targets. The qualification irradiations took place in summer 2012 in a high flux in-core position of the HFR. Typical irradiation parameters (such as peak heat flux and total irradiation time) were chosen such that the LEU targets experienced overlapping irradiation conditions compared to the existing regular Mo-99 irradiation facilities, and a series of post irradiation tests were planned. In this way, an enveloping test case for the qualification of the LEU targets could be established.

The non-destructive PIE consisted of gamma scanning, profilometry and oxide layer thickness measurements, which were all successfully completed in 2012. The qualification report for the irradiation of the LEU target in the HFR has obtained a positive advice from the HFR Reactor Safety Committee and is currently being evaluated by the Dutch Authorities.

Further work is focused on the processing of the LEU targets at the Molybdenum Production Facility at Petten, which is operated by Covidien, as well as the securing of regulatory approvals associated with use and transport of the LEU Mo-99. The goal is to produce Mo-99 at Petten using LEU targets as soon as possible.

4 Glossary

AIPES	Association of Imaging Producers and Equipment Suppliers
APD	Automatic Power Decrease
ARCHER	Advanced High-Temperature Reactors for Cogeneration of Heat and Electricity R&D
DG	Directorate General
dpa	displacements per atom
EC	European Commission
EU	European Union
FAIRFUELS	Fabrication, Irradiation and Reprocessing of FUELS and target for transmutation
FP	Framework Programme
F4E	Fusion for Energy (the European Union's Joint Undertaking for ITER and the development of fusion energy)
HB	Horizontal Beam Tube
HEU	High Enriched Uranium
HFR	High Flux Reactor
HRA/HYA	Hartlepool/Heysham 1 (AGR) Plants
IAEA	International Atomic Energy Agency
IET	JRC Institute for Energy and Transport, Petten (NL)
INET	Institute for Nuclear and New Energy Technology (Tsinghua University Beijing, China)
ISI	In-Service Inspection
ISO	International Organisation for Standardisation
ITER	International Thermonuclear Experimental Reactor
JRC	Joint Research Centre
LEU	Low Enriched Uranium
MA	Minor Actinides
MARIOS	Minor Actinides in Sodium-cooled Fast Reactors
NRG	Nuclear Research and consultancy Group, Petten (NL)
PELGRIMM	PELlets versus GRanulates: Irradiation, Manufacturing & Modelling
PIE	Post Irradiation Examination
RE	1 RE: amount of radioactivity causing a dose of 1 Sv if inhaled or ingested

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Abstract

The High Flux Reactor (HFR) at Petten is managed by the Institute for Energy and Transport (IET) of the European Commission's Joint Research Centre (JRC) and operated by the Nuclear Research and consultancy Group (NRG) which is also the licence holder and responsible for its commercial activities. The High Flux Reactor (HFR) operates at 45 MW and is of the tank-in-pool type, light water cooled and moderated. It is one of the most powerful multi-purpose materials testing reactors in the world and one of the world's leaders in target irradiation for the production of medical radioisotopes.

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