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Advanced Fuels for Gen IV Reactors: Reprocessing and Dissolution


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INTRODUCTION

The ASGARD project was constructed to bridge the scientific societies working in the area of recycling of nuclear waste. Traditionally these areas have acted rather separately in European framework funded projects even if some projects e.g. ACSEPT originally focussed on separation issues did also have work performed in conversion issues. The main parts of the circular nuclear fuel cycle are shown in Figure 1 together with some of the major EU funded projects in relation to the ASGARD scope.



Figure 1 The ASGARD project in relation to the steps in a circular nuclear fuel cycle and in relation to other important projects in the domain.

Essentially the circular nuclear fuel cycle consist of: the spent fuel being dissolved in a suitable matrix (usually nitric acid). Then the elements to be further used in production of new nuclear fuels (the actinides) are separated from the rest of the waste by either hydrochemical or pyrochemical techniques. In ASGARD we mainly aim at the hydrochemical techniques, but a smaller part on specialised fuels are also discussed with respect to pyrochemical methods. After the separation, the product has to be converted to suitable precursors for fabrication of the new fuel whereupon the fabrication process starts. It is in this area where the major part of the work of ASGARD is focussed. In future nuclear systems it is possible to use several different types of fuels whereas today oxide fuels dominate completely. It is expected that regardless of fuel type the processes outlined in Figure 1 will be necessary and thus in ASGARD we have decided to divide the work in domains given by the fuel types and tasks relating to the specific questions arising for that particular fuel type, see Figure 2.

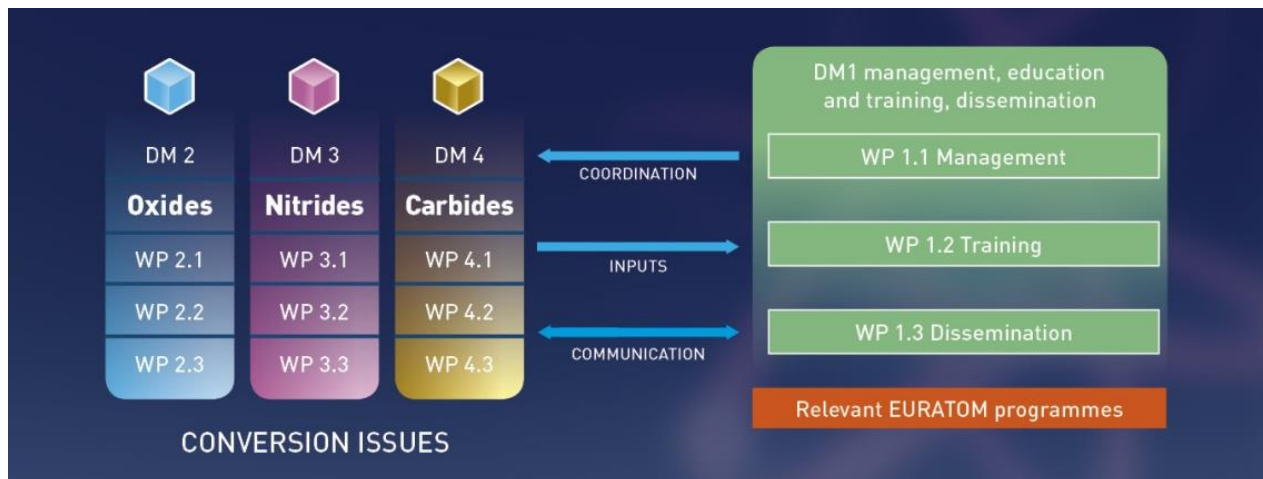


Figure 2 General structure of the ASGARD project.

As seen in Figure 2 there is a clear cross cutting activity when handling the conversion issue. There are also other specific issues that will cut cross the domains in more specific areas.

The work package division for each domain is given in Figures 3-5 below.

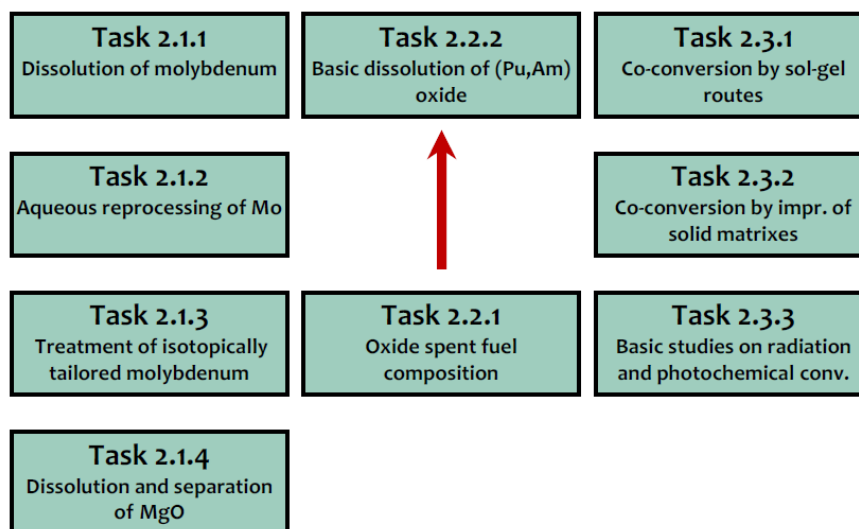


Figure 3 The organisation of the oxide domain.

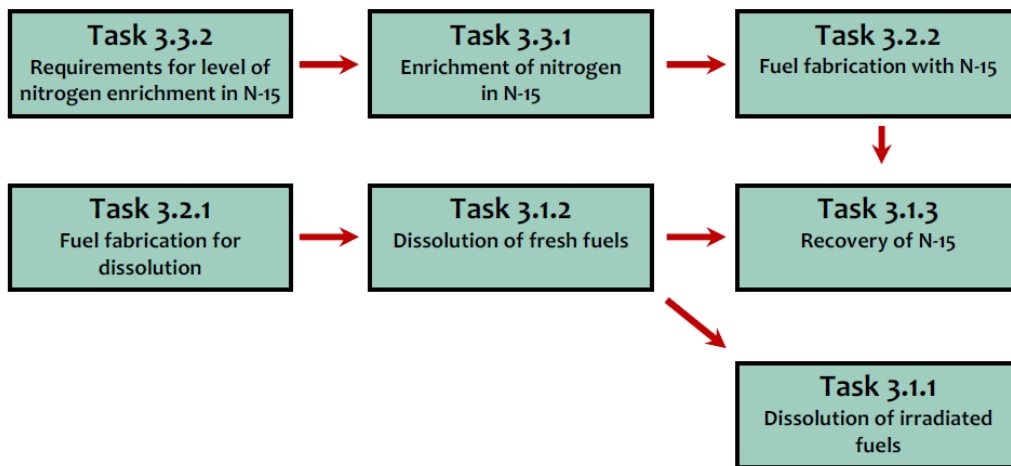


Figure 4 The organisation of the nitride domain.

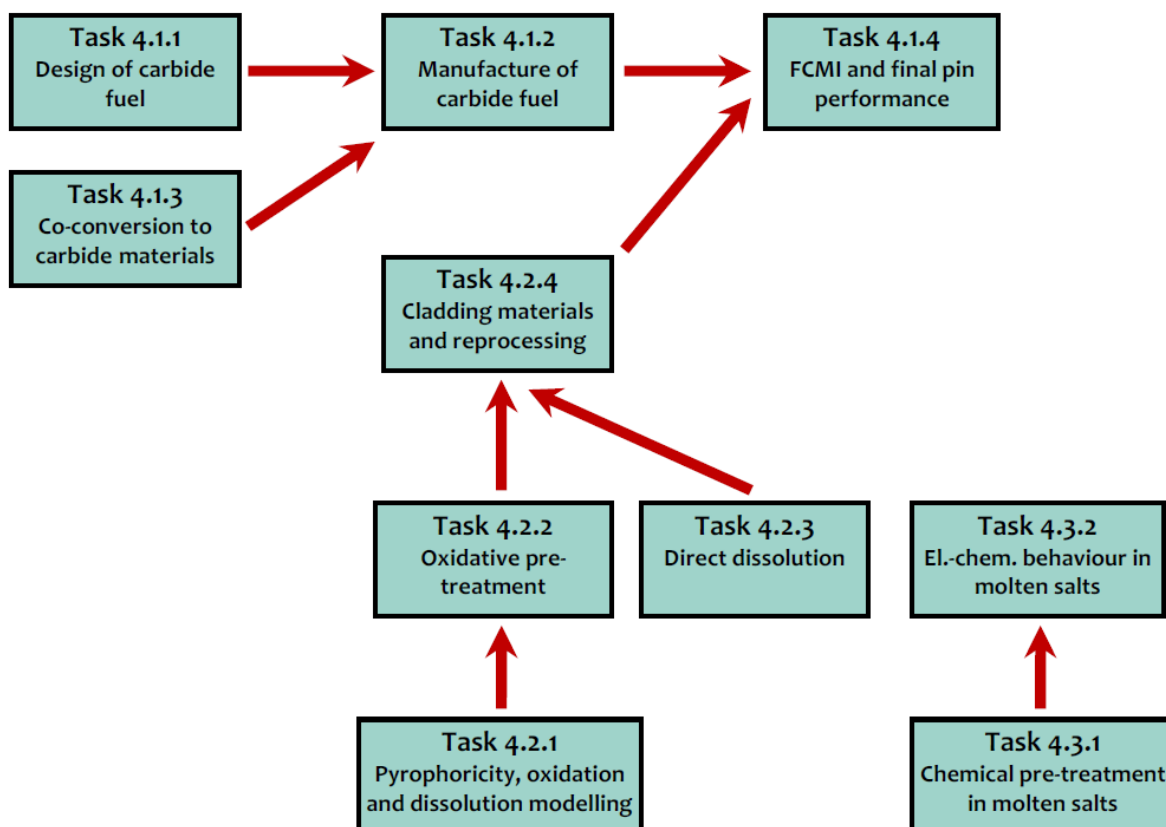


Figure 5 The organisation of the carbide domain.

Within the ASGAR project, 17 partners from 9 countries work together well distributed within the domains described above. The participation of the main nuclear actors in Europe will ascertain that the project will deliver scientific results of the highest standard in a structured and useful way. The last part is being addressed by a selected

Scientific Advisory Board and End Users Group making sure that the results of the ASGARD project are in line with what the future industry can accept at the same time as it is on a high scientific level.

Last but not least the ASGARD project has an extensive scheme for training and education of younger researchers. This program comprise not only dedicated lectures and summer(winter)schools, but also funding possibilities for shorter or longer visits to perform work in other laboratories than their own. In addition, it is also possible to receive funding to present ASGARD work at international meetings and conferences thus promoting not only the aspect of broadening the experiences of the younger researchers but also serves as an excellent tool for disseminating the results of the ASGARD project.

1 PROJECT SUMMARY FOR THE THIRD YEAR

The third year of operation of ASGARD has ended and we now face the last year. However, due to the unexpected shut down at one of the important hot labs for several months during the second year of ASGARD it is most likely that we will apply for a project extension of six months. It is, however, very nice to observe that even though some work could not be performed during the several month shut down of NRG most of the deliverables have still been met and only a few has been affected. This has been due to an effective management by the project coordination committee as well as a willingness of several partners to be flexible in when and how the work is performed.

As we now reach the last year of operation of ASGARD it is clear that some of the set out goals have been reached. The goals were typically both how to perform a good fuel pellet but also to investigate its recycling potential and identify and possibly investigate specific problems for specific cases. This will allow for a good knowledge based possible down selection of fuel types to be investigated further in the future in order to finally suggest one or two that are the most likely to use from a perspective of manufacturing and recycling

For the DM1 we have had one Winter School has been held in Stockholm in January 2014 having as subjects fuel characterization and isotope separation. In addition, a continuous feed-back and eventual improvements with regard to safety and handling of materials will be established and implemented throughout the project. The key learning points will be collected, documented and presented at the end of the project. Further collaboration with other projects such as SACESS will be undertaken in the future showing a clear added value for all partners included and at the same time create a good atmosphere and contacts especially in the younger scientist community across Europe.

A successful travel and mobility support have been developed and implemented in ASGARD under DM1 named Travel Fund, aimed at allowing young scientists, students and trainers to disseminate and network into the nuclear community, as well as allowing access to relevant facilities. So far, 20 grants have been approved, of which 4 were for mobility to other laboratories, 1 for a trainer mobility and 15 for summer/winter school and conference participation.

Outreach of ASGARD project is measured by DM1 in numerous publications in peer-review journals, as well as conferences and public media. To the date, a series of at least 14 press releases and 52 conference contributions and publications are achieved so far

In the scientific domains a considerable amount of work has been performed in all the directions of ASGARD and a more detailed account of these are given in the later sections of this report. However, some short, selected results are given here.

For domain 2 the fuel fabrication and dissolution of both CERCER and CERMET fuel have been performed. The understanding of the effect of molybdenum during dissolution (in Mo containing CERMET fuel) has been increased by detailed chemical experiments on molybdenum chemistry as well as practical enhancements in the dissolution by adding e.g. iron to the dissolution liquor. Similar experiments were performed for magnesium containing CERCER fuel where the dissolution of fabricated pellets have shown details on the kinetics and how the pellets are affected by different leaching environments. Both Mo and MgO based fuels have been manufactured and their specific challenges for a possible recycling and separation scheme were investigated. The production of the high Am containing MOX was delayed but will be performed in the next period. All the planning and safety assessment for this production has been made

For nitride fuels, i.e. DM3 the progress has mainly been in 2 areas. Pellets have been produced using both the dry and the wet route. In the second case zirconium, plutonium nitrides were manufactured and investigated using the EXAFS technique. Dissolution studies have been performed successfully on uranium nitride samples and studies on plutonium material has been prepared and will commence in early 2015. It was shown that it was possible to treat the nitride material with hot steam and thus produce a pure ammonia stream for the recovery of the enriched nitrogen. The production of enriched nitrogen has been advanced toward a significantly more economical and practical process. This is a promising step for the cheaper use of nitride fuels since the isotope enrichment is one of the important limitations of this kind of fuel.

In DM4 where the carbide fuel chemistry is investigated both front and back end related issues have been addressed. It is clear that carbide powder is pyrophoric and thus fabrication needs to be done under controlled atmospheres. A significant effort has been put into understanding this pyrophoricity and to make experiments investigating its behaviour. A detailed understanding of this process has been obtained through 2D and 3D computer modelling.

In the case of reprocessability two main routes have been investigated, direct dissolution and oxidative pre-treatment. For direct dissolution of uranium carbide experiments have been performed and previously unknown organic materials that are generated in solution have been identified. A clear effect on the subsequent separation process has been observed. The kinetics of carbide dissolution has been explored for actual unirradiated UC fuel pellets. For the oxidative pre-treatment route issues relating to volatility of several fission products have been observed on irradiated carbide fuel. Thermodynamic calculations have also shown that using CO₂ to oxidise (U,Pu)C can limit the amount of insoluble PuO₂ residue left when the pre-oxidised material is dissolved.

Molten salt treatment of carbide fuels has also been assessed, especially with respect to the cladding material. A literature review has been performed and it was clear that the cladding material can be oxidized in molten salts

Christian Ekberg

ASGARD Coordinator

2 DOMAIN SUMMARIES FOR THE THIRD YEAR

2.1 DOMAIN 1 - MANAGEMENT, EDUCATION AND TRAINING, DISSEMINATION

The management of the ASGARD project has proceeded as planned during this year. There were two technical meetings organized thanks to our partners from KTH and INCDT. The first one was held in January in Stockholm and the second one in July in Leeds. In both occasions we had the scheduled additional meetings of the PCC and GB. No serious issues had to be handled in either committee due to the smooth running of the project. The decision to ask for an extension of ASGARD was postponed to the first GB of 2015 in order to observe the situation regarding deliverables at that time. After the Stockholm meeting the scientific advisory submitted their report on the successful progress of ASGARD. It is pleasurable to note that their evaluation of the work performed in ASGARD was very favorable and that the project is a good track towards its final goals set at the time of application. No greater deviations in time or scope are to be expected.

The other main objective of DM1 is to stimulate exchange of knowledge and practical experience among the community and future researchers. Students (BSc, MSc and PhDs) represent the primary target group. Teachers and other members of the community will benefit from ASGARD activities and measures in area of education/training and mobility. Within the ASGARD project so far, two summer/winter schools were delivered. The first summer school (named Winterschool 2013, previously reported) was held on 29 – 30 January 2013 at NRG, Petten, the Netherlands. That course dealt with 'Fabrication methods & Irradiation performance' and was organized in cooperation with the 7TH FP projects FAIRFUELS and CINCH.

The first Training Session took place in January 2014 at the ASGARD progress meeting, held in Stockholm, Sweden. The Training session has been delivered by KTH and INCDT experts in fuel preparation and isotope separation. The session covered the theoretical foundations for enrichment of N-15 and the practical handling needed for the fabrication of nitride fuels from the raw material to the final pellet. In this Training Session 8 students participated. One important aspect was to give the participants hands on experience of working with the techniques used for fuel fabrication. The actinide material was in this case uranium. Thus this Training session was a base for the second Training Session to be held at Chalmers in the spring of 2015. In this session oxide or nitride fuels containing plutonium will be fabricated in an all glove box exercise. This session will be a joint venture between ASGARD and SACSESS.

The Travel funds previously erroneously reported as concluded is now operative again. The Project webpage <http://asgardproject.eu/> has been redesigned and launched successfully.

The First ASGARD International Seminar was organized as one of the tracks of the 17th International Radiochemical Conference (<http://www.radchem.cz/>) that took place on the 11–16th May 2014.

To the date, a series of at least 14 press releases and 52 conference contributions and publications are achieved so far.

2.2 DOMAIN 2 – OXIDE FUELS

The aim for DM2 is to investigate the properties of inert matrix fuels such as CERMET (AnO, Mo), CERCER (MgO, AnO), and oxide fuels (AnO). As Inert Matrix Fuels contains significant amount of inert matrix (e.g. MgO, Mo) special head end treatments after dissolution are necessary. On the other hand, it is necessary to remove the inert material to simplify the separation of the actinides and to improve the final conditioning of the remaining fission products before vitrification.

Mo/CeO₂ pellets (5, 10, 25 and 40% CeO₂) were fabricated and characterised by means of density measurements, micro hardness measurements, optical microscope and SEM investigation, and XRD. CeO₂ is homogeneously distributed in the molybdenum matrix. The separation of molybdenum by thermal treatment has been investigated. Dissolution experiments on mixed Mo/CeO₂ pellets have been performed.

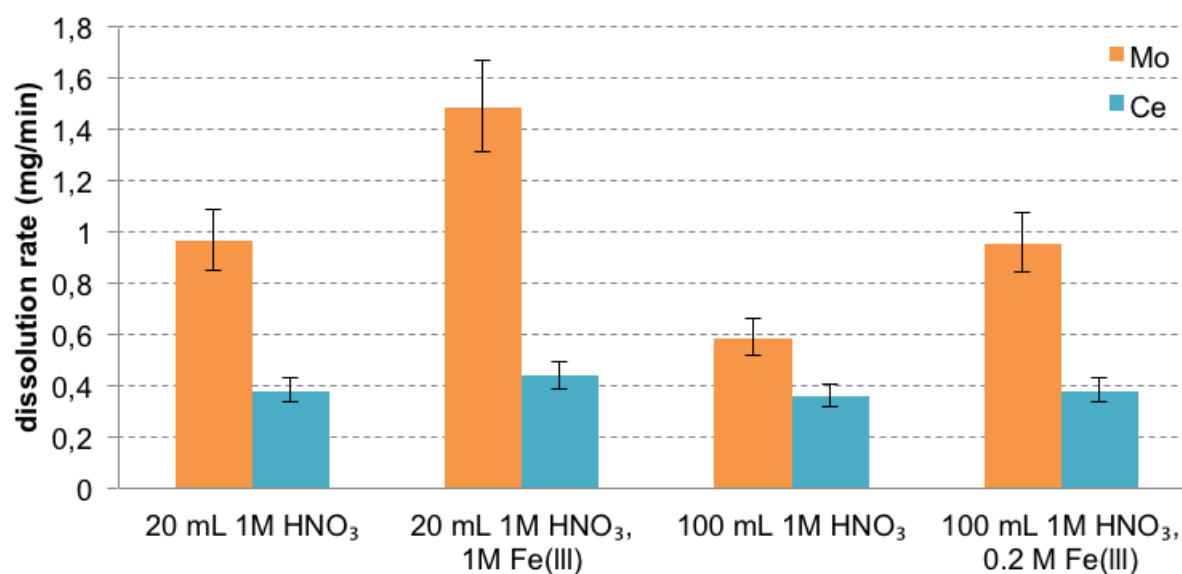


Figure 6 Dissolution experiments of CeO₂ –Mo pellets (1 g, 40 wt.-% CeO₂)

Mo/UO₂ and Mo/PuO₂ pellets were fabricated, characterised and transported to KIT for further dissolution studies. The dissolution parameters for the experiments on Mo/UO₂ and Mo/PuO₂ pellets to be performed at KIT have been discussed and pretested with Mo powders.

During the pellets fabrication, zinc stearate is used additive to provide good quality fabrication process. Thus, the dissolution solution will contain Zn(II). The extraction of Zn(II) from 0.1–3 mol/L HNO₃ into TBP, DMDOHEMA and TODGA solvents was studied. Zn(II) was shown not to be extracted in PUREX and DIAMEX type processes.

The influence of acid concentration and temperature on the dissolution rate of Mo and the influence of Fe(III) on the solubility of Mo have been investigated. The dissolution rates increase with increasing acid concentration, temperature and Fe(III) content. Fe(III) increases the overall solubility of Mo. Unfortunately, increasing temperature and nitric acid concentration leads to increased precipitation.

To clarify whether the Mo matrix forms mixed species with actinides upon dissolution in HNO₃, mixed ⁹⁸Mo and ⁹⁰Zr (IV) (as analogue for Pu (IV)) solutions have been measured by electrospray ionization mass-spectrometry (ESI-MS). The formation of mixed Mo-Zr species in nitric acid was observed. The mixed species relative abundance of Mo decreases with decreasing Zr concentration and decreasing nitric acid strength in the samples. The formation of poorly soluble mixed Mo-Zr compounds could affect the reprocessing procedure. As reported earlier, a small fraction of molybdenum in solution is present in the oxidation state +5.

The investigation of solutions containing Mo plus Eu(III) (as Am(III) analogue) at 0.5, 1 and 3 mol/L HNO₃ were successfully performed. These show the formation of mixed Mo-Eu species such as MoO₂Eu(NO₃)(OH)₃⁺(H₂O)_n. Other mixed Mo-Eu species may have been present in solution or may have formed during the measurement. The formation of mixed Mo-actinide species might have implications for the recovery of the actinides from the fuel matrix in future reprocessing steps.

In order to obtain structural information on the solution species, two pure molybdenum samples in 0.5 and 3 mol/L HNO₃ were measured by ATR-FTIR. Spectra interpretation is currently underway. More solutions (aged solutions versus fresh solutions) at different HNO₃ concentrations will be investigated to follow the polymerization processes of the Mo species by the characteristic absorption bands of the MoO₂²⁺-moiety.

CYANEX® 600 shows potential for the purification of Mo by extraction of Mo(VI). Further studies will be performed regarding the masking of Zr(IV) and Pd(II), determination of loading capacity (i.e. how much Mo(VI) can be extracted into the organic phase?) and the possible co-extraction of actinide species. Also does the behaviour of Fe(III) need further consideration in case a dissolution process utilising Fe(III) has to be dealt with.

Work on the possibility to separate strontium from molybdate solution has made the most progress. Fourteen absorbents were tested – 8 commercial and 6 in-house developmental ones. Ba(Ca)SO₄ was identified as the most prospective and it was selected for future testing. A weight distribution ratio of D_g > 250,000 mL/g was found

for this material, which is a value suitable for the design of a process for quantitative separation of Sr from the concentrated solution of molybdenum. $\text{Ba}(\text{Ca})\text{SO}_4$ was used for the preparation of a composite absorber with polyacrylonitrile binding matrix. In two dynamic column experiments, it was shown that the $\text{Ba}(\text{Ca})\text{SO}_4$ -PAN absorber is very efficient for the removal of strontium from simulated molybdate solution. The practical sorption capacity of the absorber is higher than 12 mmol/g; it was possible to treat 1500 BV of the solution with the breakthrough lower than 0.1 %, and 2700 BV of the solution with the breakthrough lower than 1 %.

The repeated dynamic experiments with Cs uptake from molybdate solution onto the KNiFC-PAN absorber practically excluded one of the potential reasons of the non-symmetric shape of the breakthrough curves – too high flow rate. Hence, molecular sorption of caesium molybdate remains to be the most probable reason of this phenomenon.

HELIOS pin 5 ($(\text{Pu}_{0.8}\text{Am}_{0.2})\text{O}_2$ in Mo) dissolution has been carried out and is now considered as finished. Four pellets were dissolved. The Mo matrix was dissolved in 4 mol/L HNO_3 at ambient temperature. The dissolved material was removed. Dissolution of the actinide oxide material in boiling 8 mol/L HNO_3 with addition of HF or Ag(II) was not fully successful; a black residue remained.



Figure 7 Dissolution setup for irradiated fuel built in the Hot Cells

The procedure for preparation of composite beads of MgO -PAN and MoO -PAN and their drying for the study of their infiltration by actinides solutions has been setup. The procedure for preparation of composite beads containing a mixture of $\text{MgO}+\text{CeO}_2$ or $\text{Mo}+\text{CeO}_2$ in a polyacrylonitrile matrix has been tested. An usual beads incineration procedure (air 600°C) yields particles with insufficient mechanical strength.

The work on the dissolution behaviour of MgO is completed. The separation of MgO from the actinides can be achieved during dissolution. Agitation speed has no effect on dissolution rate, indicating that the dissolution rate is controlled by the dissolution reaction. The acid volume has no effect on dissolution rate.

TG-DSC measurements in air to study the thermal behaviour of the loaded resins (Amberlite IRC-86 and Lewatit TP-207) were performed. Moreover a thermal treatment of the particles at different temperatures in air was done and the products were characterized by the use of SEM/EDX and XRD techniques.

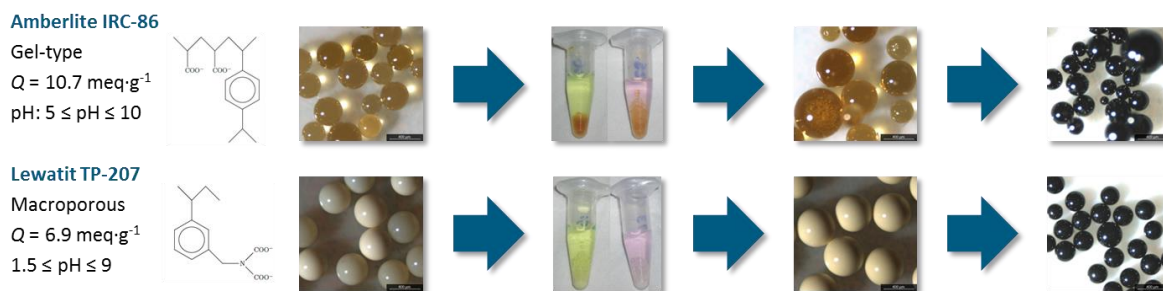


Figure 8 Overview of the process of loading resins method

For the sol gel fabrication methods, a detailed analysis of the previously generated data was done. The review of published literature data and the comparison with the experimental results lead to the conclusion that the phase composition of the $\text{Nd}_y\text{U}_{1-y}\text{O}_2$ strongly depend on the synthesis method. In the case of fabrication by powder mixing, typically two phases are formed, while the use of the sol-gel method via internal gelation leads to a single phase ceramic after a thermal treatment at 1600°C . Based on the experimental data the oxygen vacancy size was accurately determined. . The uranium(IV) / oxygen vacancy model with corrected oxygen vacancy size ($r_{\text{ov}} = 1.11 \text{ \AA}$) and the uranium(IV) / uranium(V) model are suitable to describe the lattice parameter as a function of the neodymium content. A more accurate description can be applied, taking the lattice distortion into account.

The sol-gel preparation methods focused on chemistry, characterization and optimization all steps to obtain final oxide microspheres. Increasing of amount of neodymium up to 40% in microspheres of gels causes increasing problems in the thermal treatment reduction step. The reduction to $(\text{U},\text{Nd})\text{O}_2$ caused partial destruction of spherical particles. Slower heating rate ($0.5^\circ\text{C}/\text{min}$) and longer heating (4h) in max temperature (700°C) was applied. EDS mapping analysis confirmed homogeneity distribution of all elements U and Nd (even 40%) in whole volume of microsphere. Results of X-ray fluorescence (XRF) analysis confirmed UO_2 structure with neodymium built-in in the structure of UO_2 .

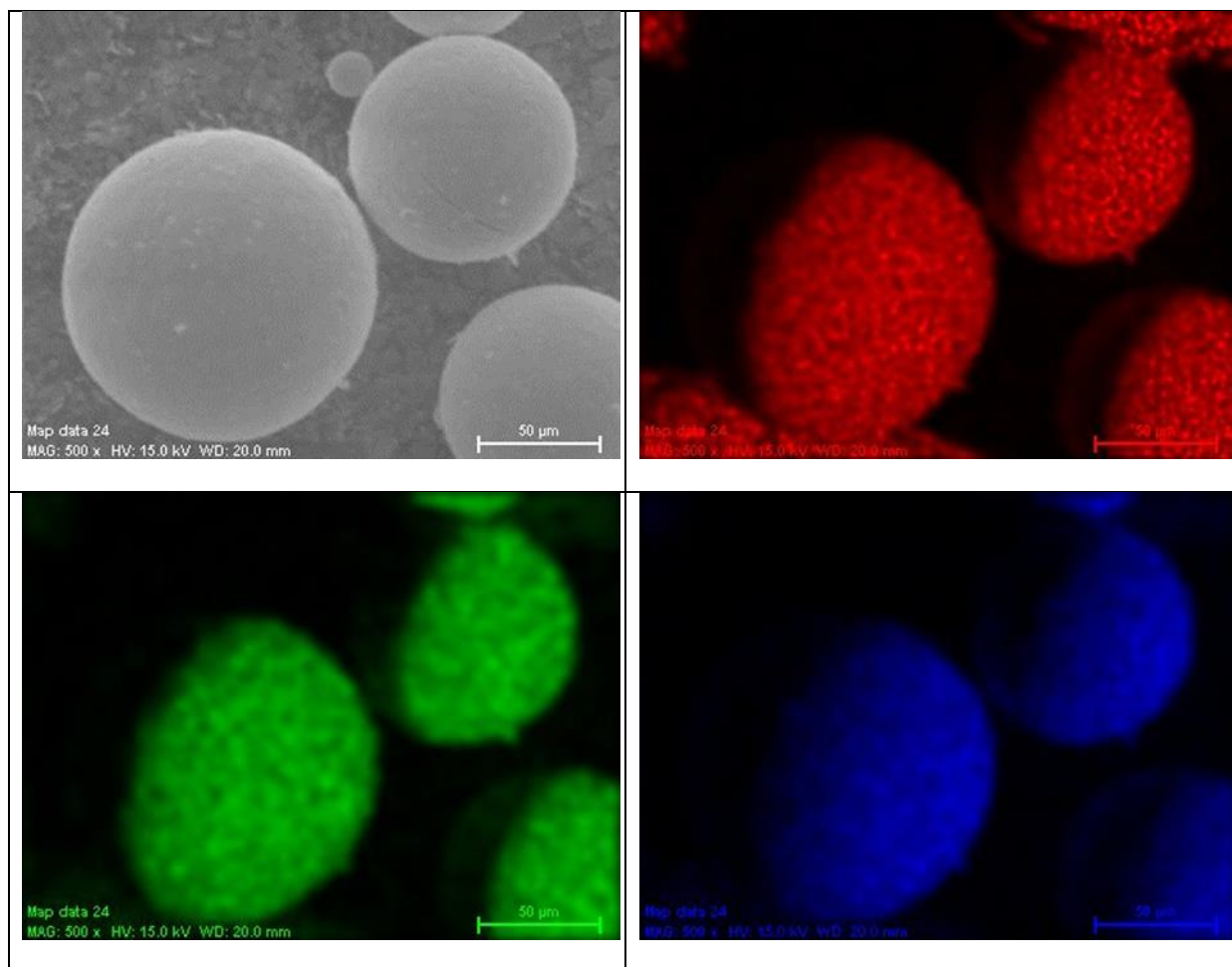


Figure 9 EDS mapping analysis of distribution elements in reduced microspheres (U,Nd)O₂ obtained by Double Extraction Process: U (red), Nd (green) and O (blue)

Considering the work dealing with UV assisted precipitation of uranium and thorium hydroxides a clear progress has been made. Material has been obtained using both UV and gamma assisted precipitation and the precipitates were investigated using both EXAFS and XRD. These material was also used to fabricate pellets. Sintering (at 1300-1600 °C) and characterizing (XRD, porosimetry, SEM) was performed. No binder or lubricant was mixed with the pellet powder (only stearic acid was used as a die-wall lubricant) and sintered pellets reached a 90-97 % TD.

2.3 DOMAIN 3 – NITRIDE FUELS

The fabrication and recyclability of nitride fuels have some main questions to address. Among these are to select the fabrication method. In ASGARD two different methods are handled: a traditional carbothermic nitridation fusing a sol-gel route and a more novel metal hydration/nitridation route.

The dissolution of irradiated CONFIRM fuel pellets has been completed. In 8 M boiling nitric acid, dissolution proceeded from the center of the pellet, leaving a residue at the (high burn-up) rim of the pellet. The residue may include a zirconia phase, but analysis is yet to be done.

Dissolution tests of fresh (Pu,Zr)N powders showed that at 110°C, full dissolution is achieved within 8 hours for 4-10 M nitric acid concentrations. At 80°C, 10 M nitric acid is required. Dissolution test of UN samples with low (less than 1000 ppm C and O) and high impurity levels confirmed that dissolution is rapid in 8M nitric acid even at room temperature. All samples dissolved completely within three hours, see Figure 10. The dissolution rate appeared to be more dependent on porosity than on concentration of C or O.



Figure 10, UN samples fully dissolved in 8M nitric acid at room temperature. Left: Sample with high C and O content. Middle: Sample with high O and low C content. Right: Sample with low O and low C content

A process for recovery of N-15 following conversion of UN to an oxide by exposure to steam at 500°C has been verified. The end product is a well defined stream of ammonia and a dry uranium oxide powder suitable for dissolution in nitric acid.

Manufacture of UN with oxygen impurities ranging from 850 to 6000 ppm weight as accomplished by controlling the time that powders are exposed to air. Using spark

plasma sintering at 1450°C, pellets fabricated from powders with low oxygen impurity reached a density of 92% TD, see Figure 11.

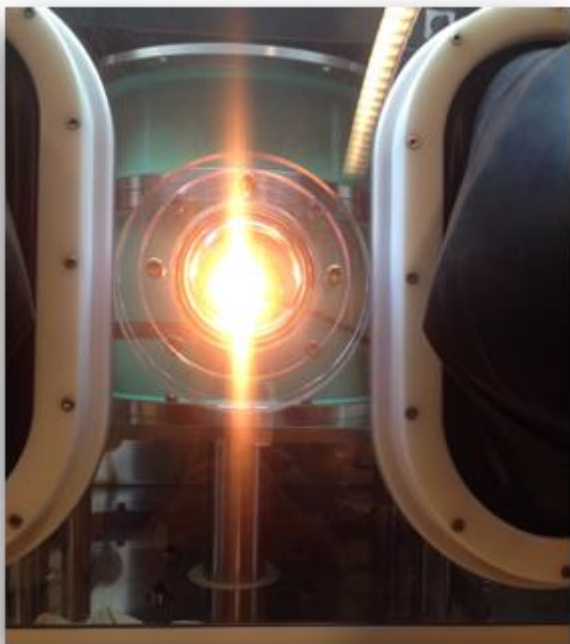


Figure 11, Spark plasma sintering of UN in a glove box at $T = 1650^{\circ}\text{C}$ resulted in pellets with zero open and less than 3% closed porosity

Zirconium, plutonium nitride fuel pellets have been manufactured using a sol-gel method. They have also been characterised using EXAFS but due to the close proximity of the signals for zirconium and plutonium(IV) no detailed data could be obtained except that it was indeed plutonium in the oxidation state +IV. No other oxidation states could be observed. A more detailed characterisation using partially destructive means will be done during 2015 including element analysis, SEM and XRD.

In order to reduce costs for N-15 enrichment, it has been shown that the flow rate of nitric acid in the column for N-15 separation can be increased by 50% by operating at a pressure of 1.2 bar. Moreover, a conversion rate of 58% for reduction of sulfuric acid to sulfur dioxide has been accomplished in an Incolloy 800 reactor by operating at 850°C.

2.4 DOMAIN 4 – CARBIDE FUELS

As with nitride fuels there are some more or less well identified issues with the production and recyclability of carbide fuels. These include the possible pyrophoricity of the carbide material itself before and during manufacture. Thus, in order to ensure a safe and secure processing envelope this pyrophoric behavior needs to be understood in detail to find possible methods for avoiding the risk.

When it comes to the recycling carbide fuel it has been suggested that there are mainly two routes: direct dissolution or a pre oxidation step followed by dissolution. Both have their pros and cons but the details are not well known. In order to select the most optimal route a detailed knowledge on the different processes are needed.

Carbide fuel pellet design modelling has defined the dimensions and creep stress enhancement material levels resulting in a pellet that is expected to exhibit reduced fuel swelling compared to a reference material by about 50% by controlling the in reactor temperature profile of the fuel pin. FCMI is also expected to be reduced as a consequence of creep rate enhancers designed into the fuel. Sphere-pac fuel size fractions that result in reduced swelling properties are defined and will be fabricated using microwave internal gelation techniques and the equipment for these tasks is being commissioned.

Methods of producing carbide materials that reduce the temperature of carbothermic reduction and minimise the need for milling and grinding have resulted in promising results, one of which has been submitted for a patent. The new method reduced the carbothermic reduction temperature to 1450°C (cf >1700°C previously). Other novel methods for the production of uranium carbide material have been demonstrated but require further carbon stoichiometry control. The compatibility of uranium carbide with barrier material for gas cooled fast reactors has been demonstrated to 2000°C using high temperature mass spectrometry experiments and microscopy analysis of in-contact UC and TiC components.

Studies on the pyrophoricity of finely divided uranium carbide powders have identified key reaction steps and strong exotherms that decrease as the oxygen concentration is lowered. The influence of powder bed size has shown the importance of mass and heat transfer effects and provided recommendations for the safe handling of carbide powders, see Figure). Models have been developed to simulate powder reactivity and reactivity of dense pellets, including diffusion through gas and solid layers produced as the reaction proceeds.

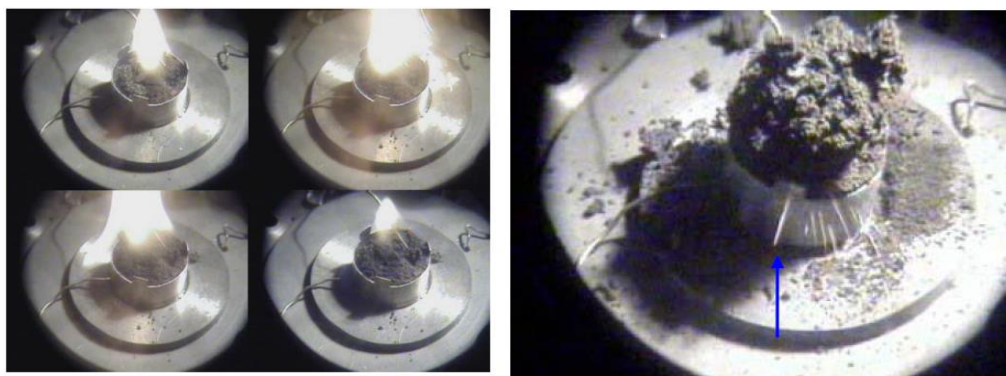


Figure 12. Ignition near 100°C of 6 mm thick UC powder bed (3 grams)

One option for the reprocessing of (U,Pu)C fuels is pre-oxidation of the carbide followed by dissolution in nitric acid, but the oxygen potential needs to be carefully controlled to avoid the generation of insoluble plutonium rich phases. The oxidation of (U,Pu)C in CO₂ atmosphere has been demonstrated and shown to conform to thermodynamic calculations. Dissolution tests of the resultant (U,Pu)O₂ material show lower levels of insoluble PuO₂ residues compared to MOx dissolution and the lack of highly colored organics compared with direct dissolution of (U,Pu)C. High temperature heating of irradiated (U,Pu)C fuel has shown how volatile fission products behave under oxidizing atmospheres and allow for some quantification of the release fractions from the fuel.

The direct dissolution of carbide fuels in nitric acid is another possible reprocessing route. The generation of CO₂ and soluble organic material that can interfere with solvent extraction efficiencies has been verified. The kinetics of dissolution of titanium carbide barrier material have shown similar partitioning of carbon between the gas and liquid phases with the production of insoluble species at high nitric acid concentrations (> 12M). Dissolution studies on arc melted ingots (1g) and breeder fuel pellets of unirradiated uranium carbide (70g) has been performed over a wide range of experimental conditions (50 – 110°C; 6 – 14M HNO₃), see Figure 13. The identification of previously unidentified soluble organic compounds has provided a breakthrough in understanding how to destroy the problematic organics. An understanding of the importance of nitrous acid upon pellet dissolution is confirmed. Solids that have precipitated during dissolution have been identified and their impact on an industrial scale dissolution process is being considered.

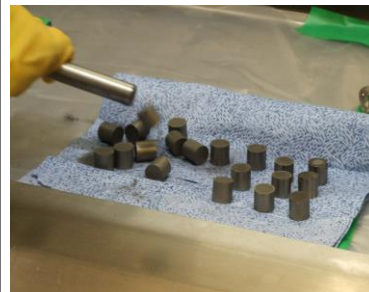
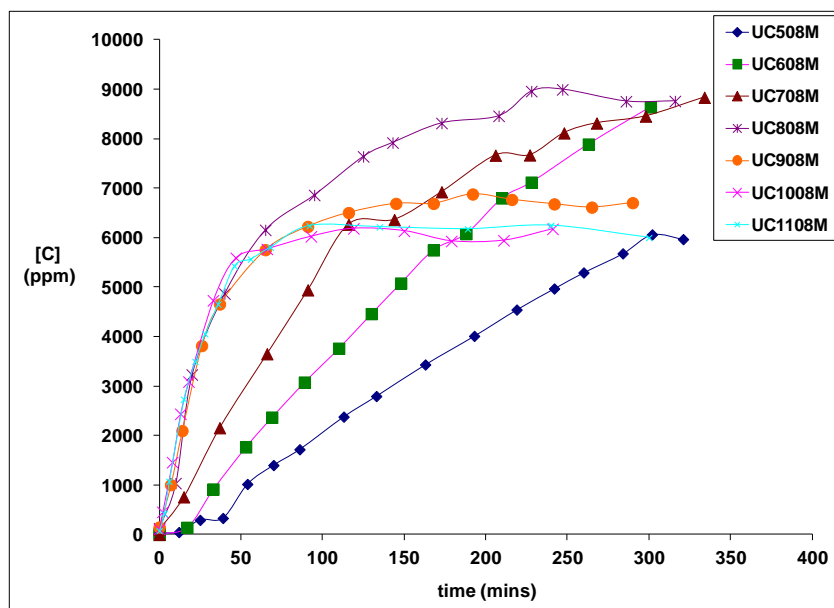


Figure 13. Influence of temperature on the total carbon dissolved throughout the 70g UC pellet (right) dissolution. $[\text{HNO}_3]_{\text{ini}} = 8\text{M}$. (e.g. UC50 is 50°C and UC 110 = 110°C)

Molten salt processing of carbides is an alternative method to oxidative pre-treatment and has been demonstrate simply using molten salt + oxygen donor materials and the conversion of SiC and TiC to oxides has been demonstrated. The same reaction is less successful on ceramic SiC samples.

CONCLUSIONS

As seen from the accounts for the activities given above the ASGARD project is well in line with the expectations at this stage. The training and mobility scheme is working well and resulting in several joint publications between different laboratories. This shows the success of the integration of the partners into the total goals of the project. Such integration is further enhanced by the inter laboratory visits promoting not only the success of ASGARD itself but also the personal development of the young scientists included.

The delays previously encountered due to an unexpected shut down of one of the facilities have been handled successfully resulting in only minor delays of the planned work. It is expected that these delays may cause a need for an extension of the project by some months but not more.

Technically the work is now proceeding with the pace expected considering the experienced partners included. Significant progress has been made in all domains showing that all the material suggested from the beginning has been manufactured. However, it need to be noted that in some cases this manufacture is a first batch only and significant further work is needed to produce material suited for further recycling studies to be relevant.

Dissolution of previously irradiated material has also been done creating a clear added value of connecting and collaborating with previous EU funded projects. This is especially important for the continuous collection of information, education of the new generation of scientists and last but not least that work previously performed is not simply forgotten and not used to its full potential.