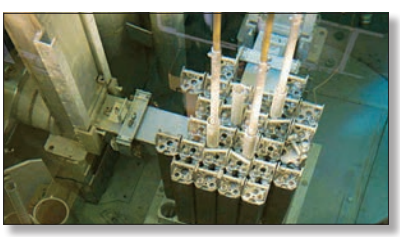
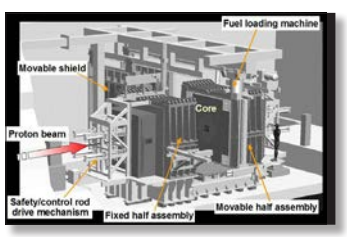
Nuclear Science
2015

Review of Integral Experiments for Minor Actinide Management





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Nuclear Science

Review of Integral Experiments for Minor Actinide Management

NEA No. 7222

NUCLEAR ENERGY AGENCY
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The mission of the NEA is:

- to assist its member countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes;
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The NEA Data Bank provides nuclear data and computer program services for participating countries. In these and related tasks, the NEA works in close collaboration with the International Atomic Energy Agency in Vienna, with which it has a Co-operation Agreement, as well as with other international organisations in the nuclear field.

Foreword

Spent nuclear fuel contains minor actinides (MAs) such as neptunium, americium and curium, which require careful management. This becomes even more important when mixed oxide (MOX) fuel is being used on a large scale since more MAs will accumulate in the spent fuel. One way to manage these MAs is to transmute them in nuclear reactors, such as light water reactors, fast reactors or accelerator-driven subcritical systems.

The transmutation of MAs is, however, not straightforward, as the loading of MAs generally affects physics parameters, such as coolant void, Doppler and burn-up reactivity. In addition, due to a lack of measured MA data, and by extension accurate MA nuclear data, it remains difficult to establish a detailed design of transmutation systems with the capacity to precisely predict the composition of the spent fuel. Nuclear data of the major actinides, such as ^{235}U, ^{238}U and ^{239}Pu, are based on a vast number of differential and integral experiments, using accelerators, critical facilities and experimental reactors. Integral experiments on MAs are much more difficult because of material handling restrictions at facilities, the difficulty of sample preparation and the need to improve measurement techniques to reduce the influence of background radiation. Moreover, most facilities for nuclear data measurements and validation are older facilities.

Therefore, the NEA Nuclear Science Committee (NSC) decided to establish an Expert Group on Integral Experiments for Minor Actinide Management (EGIEMAM) to critically review integral experiments, recommend additional integral experiments for validating MA nuclear data and investigate the potential for establishing an international framework with a view to facilitating integral experiments for MA management. The EGIEMAN has recommended integral measurements of several minor actinide nuclides from the viewpoint of the design of transmutation systems and of fuel cycles. New actions have been proposed for reactor physics and irradiation experiments, and recommendations have been made for an international comparison of the analysis of integral experiments for minor actinides.

The expert group has worked closely with the Working Party on Scientific Issues of Reactor Systems (WPRS), the Working Party on Scientific Issues of the Fuel Cycle (WPFC) and the Working Party on International Nuclear Data Evaluation Co-operation (WPEC). The relevant activities of these working parties include minor actinide burning in thermal reactors (MA-ThR), the International Reactor Physics Experiment Evaluation (IRPhE) Project, a comparative study on homogeneous and heterogeneous recycling of transuranics (TRU) and on potential benefits of advanced fuel cycles with partitioning and transmutation, meeting nuclear data needs for advanced reactors and the combined use of integral experiments and covariance data.

This report presents the main results of the discussions in the EGIEMAM, focusing on the requirements of nuclear data for MA management, the specification of important nuclear data for MA management, the review of existing integral data, the identification of bottlenecks and possible solutions, as well as the recommendation of an action programme for international co-operation.

Acknowledgements

The NEA Secretariat would like to acknowledge the contributions of members of the NEA Expert Group on Integral Experiments for Minor Actinide Management. In particular, work completed by the following experts for this report is greatly appreciated: C.S. Gil (KAERI), G. Glinatsis (ENEA), K. Hesketh (NNL), O. Iwamoto (JAEA), R. Jacqmin (CEA), Y. Khomyakov (IPPE), A. Kochetkov (SCK•CEN), M. Kormilitsyn (RIAR), S. Okajima (JAEA), G. Palmiotti (INL), G. Perret (PSI), A. Rineiski (KIT), V. Romanello (KIT), M. Salvatores (INL), D. Sweet (AMEC) and K. Tsujimoto (JAEA).

This report is dedicated to the memory of Dr Richard D. McKnight, who passed away on 28 August 2013. Dr McKnight first came to Argonne National Laboratory (ANL) in the early 1970s to work on his PhD. An international expert in the field of reactor physics, criticality safety and nuclear data, Dr McKnight was well-known at the NEA for his enthusiastic and highly valued contributions in all of these technical areas over several decades. The authors and the NEA Secretariat would like to recognise Dr McKnight for his outstanding contributions and support.

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Executive summary

It is essential to establish a reliable fuel cycle and safely manage radioactive waste when pursuing the sustainable use of nuclear fission energy. One of the key aspects involved in realising this goal is the appropriate management of minor actinides (MAs). An effective way to manage MAs is to transmute them in nuclear systems. However, due to a lack of experimental data, which has meant an absence of precision in MA nuclear data, it remains difficult to establish a detailed design of transmutation systems with reliable accuracy and the capacity to precisely predict the composition of spent fuel. The NEA Nuclear Science Committee (NSC) was therefore asked to undertake a critical review of integral experiments to validate MA nuclear data with the appropriate quality assurance for MA transmutation in nuclear systems. The NSC established the Expert Group on Integral Experiments for Minor Actinide Management (EGIEMAM) in 2009.

The EGIEMAM comprised 19 members from 10 countries. Seven meetings were held between September 2009 and September 2012. The expert group focused its discussions on the following subjects:

- requirements of nuclear data for MA management;
- review of existing integral data and identification of specifications concerning required experimental work;
- identification of bottlenecks and consideration of possible solutions;
- proposal of an action programme for international co-operation;

This report presents the results of the discussions in the EGIEMAM.

Requirements of nuclear data for MA management

First, the state-of-the-art nuclear data for MA management was surveyed and the evaluated data and the covariance data for fission and capture cross-sections of MAs were compared to recognise the uncertainties of the cross-sections.

Second, relevant activities to MA management in international organisations, such as the OECD Nuclear Energy Agency (NEA) and the International Atomic Energy Agency (IAEA), were reviewed.

Third, the expert group identified important nuclear data for MA management from the viewpoints of reactor design and fuel cycle in accordance with the results of the above mentioned review.

Status of MA nuclear data

For the fission cross-sections, as a whole, the evaluated cross-sections agree with each other within the uncertainty over most of the energy region. When the cross-section data above the fission threshold energies around 1 MeV for ^{237}Np, ^{241}Am and ^{243}Am are compared with the latest evaluated nuclear data libraries, a few percent differences in cross-sections are found. These differences are consistent with the values of covariance data. In the vicinity of the threshold energies, the evaluated uncertainty and cross-section deviations tend to increase up to several tens of percent.

As an example, for the capture cross-sections of ^{241}Am, the differences between the recent nuclear data libraries are around 10% in the neutron energy below 300 keV. Above the neutron energy of 300 keV, the deviations become larger because the cross-sections decrease and the experimental data show larger uncertainties. In addition, theoretical model prediction accuracy becomes lower for higher energies due to uncertainties from modelling competitive reactions. The situations of ^{237}Np and ^{243}Am are similar to that of ^{241}Am. Other MA capture cross-sections, for which no experimental data exist, have much larger uncertainties.

Review of internationally relevant activities

The NEA Nuclear Science Committee Working Party on Evaluation Co-operation (WPEC), the Working Party on Scientific Issues of the Fuel Cycle (WPFC), and the Working Party on Scientific Issues of Reactor Systems (WPRS) established subgroups (SGs) and/or expert groups (EGs) to discuss the main topics related to minor actinide management.

In the WPEC, subgroup 26 (SG26) performed a comprehensive sensitivity and uncertainty study to evaluate the impact of neutron cross-section uncertainty on the most significant integral parameters related to the core and fuel cycle of a wide range of innovative systems. They defined many target accuracies for different reactions, isotopes and energy ranges to meet design requirement uncertainty for many integral neutronic parameters. The subgroup 33 (SG33) defined a practice exercise to test different issues in the adjustment methodology. This consisted in selecting a limited number of available well-defined integral experiments and a list of isotopes for which cross-sections are to be adjusted based on the observed discrepancies between calculation and experimental results. Finally, recommendations for a set of best and consistent practices were made in order to improve evaluated nuclear data files. However, no experiment or minor actinide isotope was included.

The Expert Group on Fuel Cycle Transition Scenario Studies in the WPFC reviewed the R&D needs and relevant technology that would enable an efficient transition from current to future advanced reactor fuel cycles. They assembled and organised institutional, technical and economic information critical to the understanding of the issues involved in the transition from current fuel cycles to long-term sustainable fuel cycles or a phase-out of the nuclear enterprise. They also provided a framework for assessing specific national needs related to that transition.

The expert group released a publication entitled *Nuclear Fuel Cycle Transition Scenario Studies* dealing with a world scenario study in which both uranium resources and waste production issues were addressed. In particular, a review of the energy demands was discussed in detail and an original approach was proposed. Then, a review of available uranium resources was presented. A scenario with only pressurised water reactors (PWRs) in an open-cycle scheme and another with PWRs shutdown and fast reactors (FRs) start-up led only by plutonium availability and reactor lifetime were investigated. In this latter case, natural uranium consumption versus time was selected for the once-through option and the introduction of fast reactors was assessed. Also, analysis showed that the adoption of fast critical reactors can reduce the amount of minor actinides in the fuel cycle up to a factor of six until the end of the next century. Special FR breeder libraries were developed for this task in order to properly simulate fast growing regions. The impact of cross-sections uncertainties on MA mass flows and stocks was not assessed, but it was recognised that this topic should be of interest for successive WPFC studies.

The Expert Group on Reactor Physics and Advanced Nuclear Systems (EGRPANS) in the WPRS published a report on *Minor Actinide Burning in Thermal Reactors*. They concluded that the burning of minor actinides in thermal reactors was technically feasible, certainly with respect to the impact on reactor neutronics parameters, and might offer the potential to significantly reduce the quantities of some long-lived isotopes in spent fuel. However, various aspects of fuel management, particularly fabrication and handling, are

likely to present significant challenges, notably in the area of radiological protection. Also, it should be recognised that the desired reduction of radiotoxicity in the resulting waste is inherently more limited in thermal reactors than in fast reactor systems. Another activity in WPRS is the International Reactor Physics Benchmark Experiments (IRPhE) Project. The aim of the IRPhE is to provide the nuclear community with qualified benchmark data sets by collecting reactor physics experimental data from nuclear facilities worldwide. This project is useful for reviewing integral experiments.

At the IAEA, the Co-ordinated Research Project (CRP) on Minor Actinide Nuclear Reaction Data (MANREAD) started in 2007 with the aim to deliver a well-documented set of experimental data that form a reference database for evaluators of minor actinides. The MAs considered are: ^{234}U, ^{236}U, ^{237}Np, ^{238}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{242}mAm, ^{243}Am, and ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm, ^{247}Cm and ^{248}Cm.

Identification of important nuclear data for MA management

(a) Consideration of reactor design

The Subgroup 26 (SG26) showed the target accuracy requirements lists of cross-sections for MAs, which covered the energy range of interest and the range of uncertainty reduction with respect to the BOLNA covariance data. In order to obtain more general requirement lists, the EGIEMAM carried out the benchmark uncertainty analyses for the following two typical systems with three different combinations of the latest versions of nuclear data and covariance data; ENDF/B-VII with COMMARA-2.0, JENDL-4.0 and JEF-2.2:

- Sodium-cooled fast reactor (SFR), which is an Na-cooled, metal-fuelled fast reactor loaded with TRU fuel of a composition as discharged by standard PWRs (i.e. Pu/MA ratio ~10);
- ADMAB, which is a Pb-cooled nitride-fuelled fast spectrum ADS; the fuel composition has a Pu/MA ratio close to one.

The analyses with ERALIB1, which was an adjusted library generated from JEF-2.2, were also performed to understand the effect of the cross-section adjustment in the uncertainty prediction.

The benchmark analysis focused on the uncertainty of k_{eff} values due to the neutron cross-section uncertainties.

From the calculated results of k_{eff} values and their uncertainties, the following general tendency was found: both ENDF/B-VII and JEND-4.0 give similar k_{eff} values, and about 2-3% larger ones than JEF-2.2. When comparing the uncertainties of the SFR and ADMAB, the calculated results for ADMAB are larger than those for the SFR. The difference between the calculated uncertainties of the SFR and ADMAB is deduced from the difference of MA contents between both transmutation systems. Also, there is no large difference in the uncertainties between COMMARA-2.0 and JENDL-4.0, and JEF-2.2 provides larger uncertainties than the others. The uncertainties are much higher than the design requirement (300 pcm). To meet this target requirement, the uncertainties for the main contributor should be reduced. When the results between JEF-2.2 and ERALIB1 were compared, ERALIB1 was shown to drastically reduce the uncertainties, especially for the SFR. From the detail analysis of this reduction, it was concluded that the cross-section adjustment is effective for reduction of the calculation uncertainty when enough integral experiments are available. This means that integral experiments devoted to improving the knowledge of nuclear data relevant to transmutation scenarios will be very useful for achieving this goal.

The important nuclides and reactions were listed to reduce the uncertainty for both MA transmutation systems, the SFR and ADMAB.

(b) Consideration of the fuel cycle

The Expert Group on Fuel Cycle Transition Scenario Studies in the WPFC assessed the relative importance of the different heavy nuclides (in particular plutonium isotopes and minor actinides) in closed fuel cycles where critical fast reactors with various conversion ratios are implemented [10]. In this assessment, they chose a regional-type scenario, in which two groups of nations were considered:

- Group A: phasing-out group of nations, with a legacy spent fuel inventory;
- Group B: on-going nuclear policy group of nations, operating in particular a UOX-PWR fleet.

A regional group of facilities, shared by the two groups of nations, which has the final goal to transmute until the end of the present century all TRU from Group A and to stabilise the MA or TRU production (according to the specific objective) of Group B.

In order to explore a wide range of fuel cycle options, three types of fast reactors with three different conversion ratios (CR) as transmuters were considered and two different minor actinides to plutonium ratios (MA/Pu) were investigated.

To obtain decay heat, ingestion radiotoxicity, spontaneous fission rate and gamma dose rate, the decay calculations of nuclide mixtures resulting from the scenario calculations were performed.

A preliminary indication of important reactions was provided. In particular, ^{241}Am (produced by ^{241}Pu) by neutron capture produces ^{242m}Am, which decays into ^{242}Cm, which by successive neutron captures allows the formation (and accumulation, since it is hard to transmute) of ^{244}Cm. Another route is the neutron capture of ^{242}Am which produces ^{243}Am, which by capture becomes ^{244}Am, which then decays into ^{244}Cm. Fission reactions could also play a role in the nuclides removal.

From these results, it can be concluded that relevant reactions that influence the formation of critical nuclides in fuel cycles are the capture and fission of ^{241}Am, ^{242}Am, ^{243}Am, ^{244}Am, ^{242}Cm, ^{243}Cm and ^{244}Cm. The EGIEMAM carried out a preliminarily uncertainty analysis. In the analysis, only neutron cross-section uncertainties were considered and no uncertainties on other nuclear parameters were accounted for. In the case of decay heat, for example, most of the TRU decay heat comes from isotopes like ^{242}Cm, ^{244}Cm, ^{241}Am and ^{238}Pu. The decay heat uncertainty is essentially related to the evaluation of the uncertainty of the inventory of these isotopes, which is due mostly to neutron cross-section uncertainty. Moreover, the production of fission products (FP) is proportional to the fission cross-section of MA that carries significant uncertainty. A further sensitivity and uncertainty analysis would be required in order to provide quantified uncertainty contributions by isotope and reaction type.

Reviewing existing integral data

The members of the EGIEMAM reviewed the existing integral experimental information related to the MA management and classified the experiments into the following four groups:

- basic experiments using critical facilities;
- sample irradiation experiments using reactors;
- mock-up experiments;
- accelerator-reactor experiments.

Basic experiments using critical facilities

The comparison with the basic experimental data obtained at critical facilities and the calculation results have been effective for the verification of the neutron cross-section. The information about the experiments performed at 14 critical facilities of 7 countries includes (a) reaction rate ratio measurements, (b) small sample reactivity worth measurements and (c) criticality measurements.

(a) Reaction rate ratio measurements

The data would be very useful in the adjustment of the multi-group neutron cross-section library. In the adjustment method, many independent experimental data are needed to eliminate the systematic experimental errors. However, there has been no experimental data of fission reaction for ^{245}Cm. For the capture reaction, there are few experimental results, especially in the fast neutron field. This would suggest that the difficulty of the sample preparation is the bottleneck, since a few hundred mg or less of samples with high purity will be needed for the capture reaction rate measurements.

(b) Small sample reactivity worth measurements

Experimental data have been obtained at seven different critical facilities in four countries.

To improve the accuracy of the neutron cross-sections for MAs, more experimental data will be needed in the fields of fast and thermal neutron spectra. Prior to the experiment, careful consideration will be needed to eliminate or reduce this core material effect.

(c) Criticality measurements

Few experimental data are available in relation to criticality for the core with the mixture of major and minor actinides, carried out in Comet and Planet Facilities in the United States.

Sample irradiation experiments

The information of sample irradiation experiments was collected from nine reactors of six countries.

The irradiation experiments are very powerful for evaluating cross-section data and validating transmutation rates for MA. To improve the cross-section data and to effectively predict the transmutation rate, it is necessary to irradiate samples in a variety of neutron spectra. The analysis of the experiment needs detailed irradiation conditions, e.g. the reactor power history, the neutron spectrum at the irradiation point or the cooling time. Some of the experiments were lacking such data since they were carried out in the 1980s.

Mock-up experiments

There has been only one series of partial mock-up experiments for the MA loaded core that was carried out at the BFS Facility in the Institute for Physics and Power Engineering (IPPE), Russian Federation. About 10 kg of ^{237}Np-dioxide pellets were loaded in the central core region and the core characteristics (e.g. criticality, reactivity coefficients, etc.) were measured.

A calculation study [13] showed that the partial mock-up experiments with loading massive MA fuels were very effective in reducing the uncertainty of the core characteristics in the design study, even if they were only partial.

Accelerator-reactor experiments

The ADS is a subcritical reactor driven by the external neutron source generated by the accelerator beam to maintain the chain reaction. In Europe, the MUSE experiments were carried out at the MASRUCA Facility and the GUINEVERE experiments are carried out at VENUS Facility. The purpose of these experiments is to investigate the validation of the subcriticality monitoring for an ADS and the applicability of a conventional calculation system for the fast reactor. In Japan, ADS experiments have been conducted in a well-thermalised critical assembly (KUCA A-core) in order to evaluate the basic characteristics of ADS by 14 MeV neutrons generated by D-T reactions or spallation neutron generated by an FFAG proton accelerator.

Combined experiments with a fast neutron subcritical core and the spallation neutron source should, however, be considered.

Identifying bottlenecks and finding solutions

In Sections 2 and 3, the important nuclides of MA for design of transmutation systems and for fuel cycles were listed: ^{237}Np, ^{241}Am, ^{242m}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm and ^{245}Cm. To improve the cross-section data for these nuclides and to design the MA transmutation system with reliable accuracy, many additional integral data are still necessary. Members of the EGIEMAM discussed the availability of MA samples and experimental facilities in order to identify bottlenecks and to find solutions.

Availability of samples

To carry out additional experiments, adequate amounts of MA samples with high purity should be prepared. A few hundred mg or less of the samples are needed for both reaction rate ratio measurements and irradiation experiments; a few tens of grams of the sample is needed for the small sample reactivity worth measurements and several tens of kg of the samples are needed for the mock-up experiments and the criticality measurements. Members discussed the possible availability of MA materials.

The Research Institute of Atomic Reactors (RIAR) in the Russian Federation, the Actinide Chemistry and Radioactive Targets at Orsay (CACAO group – Chimie des Actinides et Cibles radio Actives à Orsay) in France and the National Isotope Development Center (NIDC) in the United States are able to supply a few tens of grams or less of the samples for the reaction rate ratio measurements, the irradiation experiments and the small sample reactivity worth measurements. On the other hand, for the mock-up experiments and the criticality measurements, it is difficult to procure massive amounts of MAs. Prior to starting the experiments, the selection of MAs and the estimation of their amounts should be considered from the viewpoint of the integral experiments. These problems and solutions should also be dealt with when handling MAs not only during experiments but also during storage and transportation from the fabrication site to the experimental site.

Availability of experimental facilities

Members surveyed the status of the experimental facilities around the world. They listed the experimental facilities/reactors for the critical facilities that could be used for the basic experiments, for the irradiation experiments and for the ADS experiments.

Summary and recommendations

The EGIEMAM reviewed the state-of-the-art minor actinides nuclear data, performed uncertainty analyses and target accuracy assessments for minor actinide management and stressed the need to improve nuclear data. Existing integral experiments were also reviewed from the viewpoint of MA management and a lack of experiments and accuracy

were identified. Because of proprietary considerations, most of the results are not fully available.

As there is only a limited number of facilities and limited expertise and resources (materials, manpower, funding), the expert group stressed the need for a concerted effort paving the way for a common experimental programme, where resources can be optimised towards improving MA nuclear data knowledge.

Additional integral experiments

The members discussed integral experiments for minor actinides and minor isotopes in major actinides (MIs). After the discussion, they recognised that integral experiments for not only MAs and MIs, but also for other nuclides including major actinides, are very important. Based on current MA nuclear data knowledge and from the viewpoint of MA management, it was agreed that the following integral experiments should be carried out:

- Reaction rate measurements in fast systems:
  - capture rate for ^{237}Np, ^{241}Am, ^{243}Am, ^{244}Cm and ^{245}Cm;
  - fission rate for ^{245}Cm.
- Small sample reactivity worth measurements:
  - necessity to accumulate data for MAs in fast systems;
  - pre-analysis (e.g. estimation of reactivity worth and sensitivity analysis) is important.
- Irradiation experiments:
  - irradiation experiments in fast systems to complement the other integral experiments;
  - analysis of existing data through international co-operation.
- Mock-up experiments:
  - mock-up experiments using massive amounts of MAs will be needed in the future;
  - start of design studies to prepare the experiments (estimation of MA fuel contents, amounts, and the effectiveness of large inventory [~kg] experiments for parametric study to simulate design cores).

Recommendation of an Action Programme for International Co-operation

In addition to the proposal of the integral experiments for MAs, the EGIEMAM recommends the following actions to be performed through the international co-operation:

- Comparisons in order to obtain high reliability for integral data and analysis results:
  - international comparison of the reaction rate measurements at common facilities;
  - international comparison of chemical analysis results with the same irradiated samples in post-irradiation examination (PIE);
  - comparison of calculation results of benchmark experiments (including the selection of benchmark experiments).
- Co-operation on differential nuclear data measurements complementary to integral experiments:
  - making a proposal to WPRS;

  - sharing new cross-section measurements and evaluations in nuclides, reactions and/or neutron energy ranges.

- Co-operation on design study for the specification of MA cores for integral experiments at a zero power facility in support of current selected target designs for MA management.

To realise these proposals, the members discussed new actions using examples from past international collaboration related to integral experiments such as the Intercomparison of reaction rate measuring techniques in MASCURCA (IRMA) campaign in 1984 and the international benchmark experiment for effective delayed neutron fraction in 1995. Finally, the EGIEMAM proposed the following actions:

- Joint design of reactor physics MA measurements in selected facilities:
  - selection of reactions and nuclides to be measured including assessment of the new measurement techniques/data, and specifications of the integral experiments including target uncertainties;
  - consideration and determination of facilities and required expertise;
  - development and co-ordination of the experimental programme (including assessments and sharing of resources and results, time schedules and costs).
- International collaboration of the irradiation programme for MAs:
  - selection of reactions and nuclides to be irradiated including assessment of the new measurement techniques/data, and specifications of the integral experiments including target uncertainties;
  - selection of irradiation facilities;
  - considering preparation of the samples for irradiation and time schedule for irradiation;
  - development and co-ordination of the irradiation programme (including assessment and sharing of resources and results, time schedules and costs).

In addition, the EGIEMAM recommended that the WPRS and WPFC consider an international comparison of the analysis of integral experiments for MAs in the selection of experiments, the creation of benchmark models, the analysis of benchmark problems and the comparison of analysis results.

1. Introduction

It is essential to establish a reliable fuel cycle and safely manage radioactive waste when pursuing the sustainable use of nuclear fission energy. One of the key aspects involved in realising this goal is the appropriate management of minor actinides (MAs), such as neptunium, americium and curium found in spent fuel. In particular, the necessity to manage MAs becomes more important when plutonium is used as MOX fuel on a large scale in power reactors, including in both light-water reactors and fast reactors, since americium and curium will accumulate in the spent fuel. One effective way to manage MAs is to transmute them in nuclear systems such as fast reactors and accelerator-driven subcritical systems. The transmutation of MAs, however, is not easy. From the viewpoint of the reactor neutronics, the loading of MAs generally affects the reactor characteristics, such as the coolant void reactivity, the Doppler coefficient and the burn-up effect.

The detailed design of transmutation systems with reliable accuracy and the precise prediction of the composition of spent fuel remain difficult due to a lack of experimental MA data, and by extension accurate MA nuclear data. It should be noted that nuclear data of the major actinides, such as ^{235}U, ^{238}U and ^{239}Pu, have been improving for more than 50 years, based on a large number of differential and integral experiments, using accelerators, critical facilities and experimental reactors.

The results of integral experiments have been used to validate the nuclear data. Recently, the covariance data in the nuclear data libraries have been introduced to estimate the uncertainty in the design study of the nuclear system. The integral experimental data together with the sensitivity analysis have become more effective in improving design accuracy. Integral experiments on MAs, however, are much more difficult than those on the major actinides because of restrictions at the facilities, the difficulty of sample preparation, the necessity to improve measurement techniques to reduce the influence of background radiation, etc.

As a result, the NEA Nuclear Science Committee (NSC) was requested to undertake a critical review of integral experiments for validating MA nuclear data with appropriate quality assurance for MA transmutation in the nuclear systems. The NSC established the Expert Group on Integral Experiments for Minor Actinide Management (EGIEMAM) in 2009.

The EGIEMAM consisted of 19 members from 10 countries. Seven meetings were held between September 2009 and September 2012. The objectives of the EGIEMAM were to review integral experiments for validating MA nuclear data, to recommend additional integral experiments and to propose an international framework to facilitate them from the MA management viewpoints. The expert group members discussed the following subjects: 1) requirements of nuclear data for MA management (including evaluation of target accuracy, comparison of uncertainty analysis results among nuclear data libraries and identification of important nuclear data), 2) review of existing integral data and identification of specifications concerning required experimental work, 3) identification of bottlenecks and consideration of possible solutions, such as availability of MA sample and experimental facilities, and 4) a proposal of an action programme for international co-operation. This report presents the results of the discussions in the EGIEMAM.

2. Requirements of nuclear data for minor actinide management

2.1 Status of MA nuclear data

New evaluations of MA nuclear data have progressed taking into account of new measurements to improve applicability to advanced reactors. For the new evaluated nuclear data libraries, such as JENDL-4.0 released in 2010 and ENDF/B-VII.1 released in 2011, the number of nuclides with covariance data is significantly increased since requirements of the covariance data have been increased to estimate uncertainties especially for fast reactors. The uncertainties of fission and capture cross-sections of ^{237}Np, ^{241}Am and ^{243}Am for fast neutrons are shown below from the viewpoint of the differences between the evaluated data and the covariances.

Experimental data for MA fission cross-sections are rather abundant compared to the other reactions. As a result, the agreement between the evaluated cross-sections is good. Figure 1 shows relative deviations of fission cross-sections of recent evaluated libraries from that of JENDL-4.0 for ^{237}Np. The shaded area indicates the standard deviation of the cross-section in JENDL-4.0. The evaluated cross-sections agree within the uncertainty in the most of the energy region. Above the fission threshold energies around 1 MeV, the covariance and the differences between the recent evaluated data indicate a few per cent for important MA such as ^{237}Np, ^{241}Am and ^{243}Am, as shown in Figures 1-3. For the sub-threshold energies, where the fission cross-section decreases significantly, the evaluated uncertainty and cross-section deviations tend to increase up to several tens of percents.

Figure 1: Relative deviation from JENDL-4.0 for ^{237}Np fission cross-section

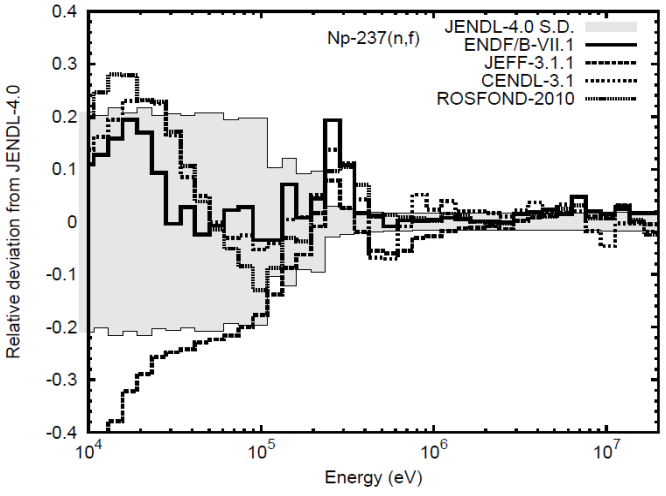


Figure 2: Relative deviation from JENDL-4.0 for ^{241}Am fission cross-section

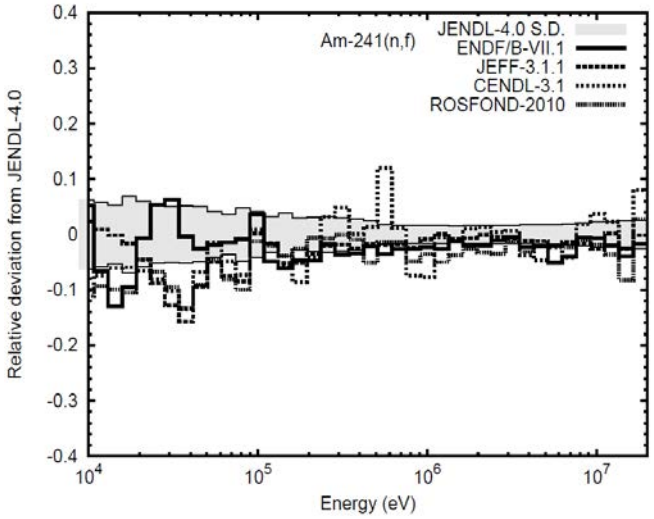
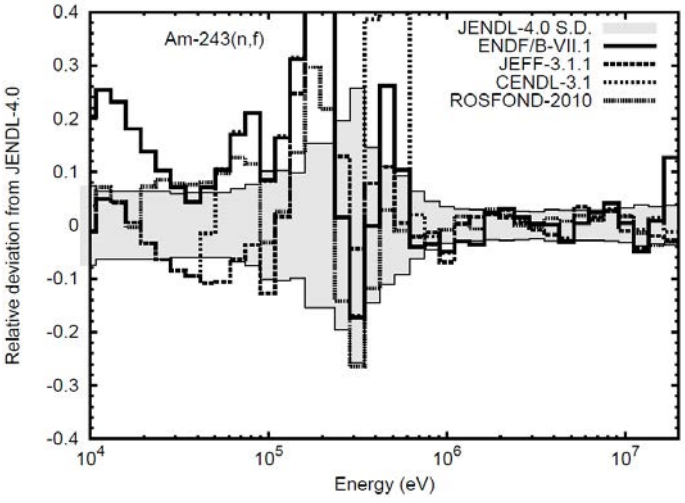


Figure 3: Relative deviation from JENDL-4.0 for ^{243}Am fission cross-section



For the capture cross-sections, experimental data are restricted to ^{237}Np, ^{241}Am and ^{243}Am for fast neutron energies. Deviations of the evaluated data from JENDL-4.0 are shown in Figure 4 for ^{241}Am. The differences between the evaluated data are around 10% below 300 keV except CENDL-3.1. Above it, the deviations become larger because the cross-sections decrease and the experimental data show larger uncertainties. In addition, theoretical model prediction accuracy becomes lower for higher energies due to uncertainties from modelling competitive reactions. The situation of ^{237}Np and ^{243}Am is similar to that of ^{241}Am, as shown in Figures 5 and 6, respectively. However, other MA capture cross-sections, for which no experimental data exist, have much larger uncertainties.

Figure 4: Relative deviation from JENDL-4.0 for ^{241}Am capture cross-section

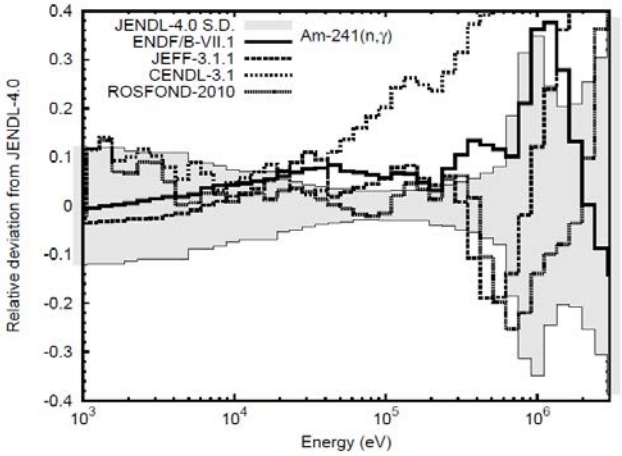


Figure 5: Relative deviation from JENDL-4.0 for ^{237}Np capture cross-section

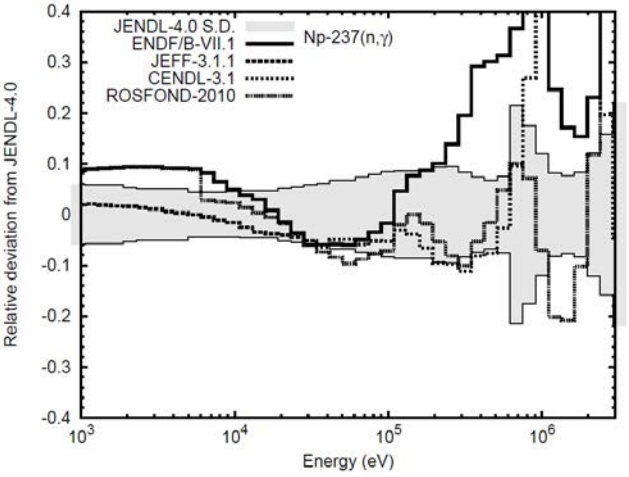
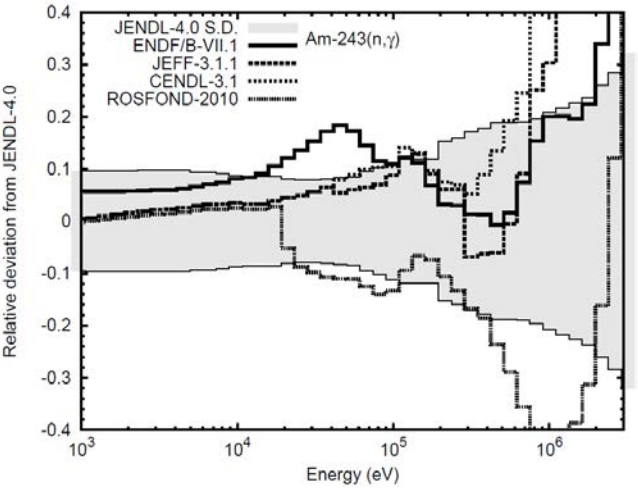


Figure 6: Relative deviation from JENDL-4.0 for ^{243}Am capture cross-section



2.2 Review of relevant activities

2.2.1 WPEC Subgroup 26

In 2005, the Working Party on International Nuclear Data Evaluation Co-operation (WPEC) established Subgroup 26 to develop a systematic approach, to define data needs for advanced reactor systems and to make a comprehensive study of such needs for Generation-IV (Gen-IV) reactors [1].

A comprehensive sensitivity and uncertainty study was performed to evaluate the impact of neutron cross-section uncertainty on the most significant integral parameters related to the core and fuel cycle of a wide range of innovative systems, even beyond the Gen-IV range of systems. Results were obtained for the Advanced Breeder Test Reactor (ABTR), the Sodium-cooled Fast Reactor (SFR), the European Fast Reactor (EFR), the Gas-cooled Fast Reactor (GFR), the Lead-cooled Fast Reactor (LFR), the Accelerator-Driven Minor Actinide Burner (ADMAB), the Very High-Temperature Reactor (VHTR) and a Pressurised Water Reactor with extended burn-up (PWR). These systems corresponded to the current studies in the Generation-IV initiative, the Global Nuclear Energy Partnership (GNEP), the Advanced Fuel Cycle Initiative (AFCI), and in advanced fuel cycle and partitioning/transmutation studies in Japan and Europe.

The covariance data used in the uncertainty analysis was the BOLNA [2] matrix, a joint effort by several laboratories contributing to Subgroup 26. The analysis mostly focused on integral parameter (like k_{eff}, reactivity coefficients, power distributions etc) uncertainty due to neutron cross-section uncertainties. The integral parameters considered were related to the reactor core performance but also to some important fuel cycle-related parameters, like the transmutation potential, the doses in a waste repository or the neutron source at fuel fabrication.

A preliminary “Design Target Accuracies” was also compiled and a target accuracy assessment was performed in order to provide an indicative quantitative evaluation of nuclear data improvement requirements by isotope, nuclear reaction and energy range that would meet the design target accuracies, as compiled in the study. The first priorities were formulated on the basis of common needs for fast reactors and, separately, thermal systems.

The Expert Group on Integral Experiments for Minor Actinide Management (EGIEMAM) has retained the SFR, a sodium fast reactor burner with ~10% MA content in the fuel), and the ADMAB, with a very high content of MA (ratio of 2:1) as the two systems of major interest. Tables 1 and 2 show the target accuracy requirements for these two systems.

These results should be used with precaution. They indicate trends and general priority needs. In fact, these quantitative values have been obtained considering only diagonal (variance) uncertainty values that represent an underestimation of the real uncertainty. Moreover, the accuracy requirements and priorities are strongly dependent on the assumed initial uncertainty variance-covariance data, in particular in the case of BOLNA matrix. More recent covariance matrix data are now available, and futher studies will be necessary to update the results.

Table 1: SFR uncertainty reduction requirements to meet integral parameter target accuracies

Isotope	Cross-section	Energy range	Uncertainty (%)	
			Initial	Required
^{241}Pu	σ_{fiss}	1.35 - 0.498 MeV	16.6	3.4
		498 - 183 keV	13.5	2.6
		183 - 67.4 keV	19.9	2.6
		24.8 - 9.12 keV	11.3	3.5
		2.04 - 0.454 keV	12.7	4.4
^{56}Fe	σ_{inel}	2.23 - 1.35 MeV	25.4	3.3
		1.35 - 0.498 MeV	16.1	3.2
^{23}Na	σ_{inel}	1.35 - 0.498 MeV	28.0	4.0
^{244}Cm	σ_{fiss}	1.35 - 0.498 MeV	50.0	5.1
^{242m}Am	σ_{fiss}	1.35 - 0.498 MeV	16.5	4.2
		498 - 183 keV	16.6	3.1
		183 - 67.4 keV	16.6	3.1
		67.4 - 24.8 keV	14.4	4.0
		24.8 - 9.12 keV	11.8	4.2
		2.04 - 0.454 keV	12.2	5.1
^{240}Pu	σ_{fiss}	1.35 - 0.498 MeV	5.8	1.8
^{240}Pu	ν	1.35 - 0.498 MeV	3.7	1.5
^{238}Pu	σ_{fiss}	2.23 - 1.35 MeV	33.8	5.6
		1.35 - 0.498 MeV	17.1	3.3
		498 - 183 keV	17.1	3.6
^{238}Pu	ν	1.35 - 0.498 MeV	7.0	2.7
^{242}Pu	σ_{fiss}	2.23 - 1.35 MeV	21.4	4.9
		1.35 - 0.498 MeV	19.0	3.5
^{245}Cm	σ_{fiss}	183 - 67.4 keV	47.5	6.7
^{242}Pu	σ_{capt}	24.8 - 9.12 keV	38.6	8.4
^{238}U	σ_{capt}	24.8 - 9.12 keV	9.4	4.3
^{56}Fe	σ_{capt}	2.04 - 0.454 keV	11.2	5.3

Table 2: ADMAB uncertainty reduction requirements to meet integral parameter target accuracies

Isotope	Cross-section	Energy range	Uncertainty (%)	
			Initial	**Required (λ=1)**
^{244}Cm	σ_{fiss}	6.07 - 2.23 MeV	31.3	3.0
		2.23 - 1.35 MeV	43.8	2.6
		1.35 - 0.498 MeV	50.0	1.5
^{56}Fe	σ_{inel}	6.07 - 2.23 MeV	7.2	2.5
		2.23 - 1.35 MeV	25.4	1.6
		1.35 - 0.498 MeV	16.1	1.5
^{243}Am	σ_{inel}	1.35 - 0.498 MeV	42.2	2.3
		498 - 183 keV	41.0	3.6
		183 - 67.4 keV	79.5	3.7
^{241}Pu	σ_{fiss}	1.35 - 0.498 MeV	16.6	2.1
		498 - 183 keV	13.5	1.7
		183 - 67.4 keV	19.9	1.7
^{241}Am	σ_{fiss}	6.07 - 2.23 MeV	11.7	1.7
		2.23 - 1.35 MeV	9.8	1.4
		1.35 - 0.498 MeV	8.3	1.2
^{245}Cm	σ_{fiss}	1.35 - 0.498 MeV	49.4	3.3
		498 - 183 keV	37.2	2.9
		183 - 67.4 keV	47.5	2.9
		67.4 - 24.8 keV	26.5	3.2
^{209}Bi	σ_{inel}	2.23 - 1.35 MeV	34.1	2.8
		1.35 - 0.498 MeV	41.8	4.2
^{243}Am	σ_{fiss}	6.07 - 2.23 MeV	11.0	2.3
		1.35 - 0.498 MeV	9.2	1.6
^{244}Cm	ν	6.07 - 2.23 MeV	11.1	2.5
		1.35 - 0.498 MeV	5.5	1.3
^{15}N	σ_{el}	1.35 - 0.498 MeV	5.0	1.2
Pb	σ_{inel}	6.07 - 2.23 MeV	5.4	2.9
^{90}Zr	σ_{inel}	6.07 - 2.23 MeV	18.0	3.3
^{238}Pu	σ_{fiss}	2.23 - 1.35 MeV	33.8	6.0
		1.35 - 0.498 MeV	17.1	3.4
		498 - 183 keV	17.1	3.9
^{242}Cm	σ_{fiss}	6.07 - 2.23 MeV	52.6	26
		498 - 183 keV	66.0	28.4
^{238}Pu	ν	1.35 - 0.498 MeV	7.0	2.8
		498 - 183 keV	7.0	3.4
^{242m}Am	σ_{fiss}	498 - 183 keV	16.6	4.8
		183 - 67.4 keV	16.6	4.8

2.2.2 WPEC Subgroup 33

As a result of the work performed in the WPEC Subgroup 26 on *Nuclear Data Needs for Advanced Reactor Systems*, several target accuracies for different reactions, isotopes, and energy ranges were defined to meet design requirement uncertainty for many integral neutronic parameters. Many of these target accuracies were very tight and not likely to be achieved with current experimental measurement techniques. It was, therefore, suggested that a combined use of integral experiments and differential information (e.g. measurements, evaluation, and uncertainty data) should be provided designers with improved nuclear data that would be able to meet design target accuracies. Then, it was proposed that, as a mandate for this new Subgroup 33, the methods and issues of the combined use of integral experiments and covariance data should be studied with the objective of recommending a set of best and consistent practices in order to improve evaluated nuclear data files.

The working method includes the following points:

- Review current methods and practices, including feedback from past work. Consider innovative approaches, as well as the methods used in non-nuclear fields when applicable. Pre-select candidate methods for testing.
- Define test cases and input data to be used in the evaluation of the pre-selected methods. For the purpose of assessing the relative merits of different approaches, the participants should agree to define:
  - the set input parameters to be adjusted, along with a priori uncertainties and correlations;
  - a set of relevant (possibly simplified) integral experiments, with their associated uncertainties and correlations;
  - the type of performance parameters, where uncertainties have to be reduced.

The integral experiments will be calculated, along with the necessary sensitivity coefficients, according to the type of parameters to be adjusted. The tests should allow a comparison of the robustness of the methods, e.g. for extrapolation, etc.

- Recommend a general methodology for data assimilation and for assessing the needs for additional experiments on the basis of the test results and analysis.

These steps were followed and a practice exercise was defined to test different issues in the adjustment methodology. This consisted in selecting a limited number of well-defined available experiments and a list of isotopes, whose cross-sections are to be adjusted based on the observed discrepancies between calculation and experimental results. However, no experiment or minor actinide isotope was included that was of relevance to the EGIEMAM.

It is possible that in the future an extended set of experiments and isotopes, which include minor actinides, could be selected.

2.2.3 WPFC

The WPFC Expert Group on Fuel Cycle Transition Scenario Studies issued a publication entitled *Transition towards a Sustainable Nuclear Fuel Cycle: A World Scenario Analysis* dealing with a world scenario study in which both uranium resources and waste production issues were addressed. First, a review of the energy demands (up to 2100, then an increase of 0.25%/year was assumed up to 2200) according to various institutes (IPCC, IAASA) was discussed in detail and an original approach was proposed. Then, a review of available uranium resources was presented based on the last NEA publications about this topic (“Red Book”); unconventional resources were considered too. A scenario with only

PWRs in an open-cycle scheme and another one with PWRs shutdown and FRs start-up led only by plutonium availability and reactors lifetime were investigated, assuming a homogeneous approach (world is seen as a single big region). Finally, the same investigation was performed by dividing the world into four big macro-regions (according to IPCC proposed scheme, i.e. OECD90, Asia, Ref (Eastern Europe), and Alm (Latina America, Africa, etc.). In this latter case, natural uranium consumption versus time was selected for the once-through option and the introduction of fast reactors was assessed, showing that only this last option can avoid resources consumption at roughly the half of the next century. The analysis also showed that the adoption of fast critical reactors can reduce the amount of minor actinides in the fuel cycle up to a factor of 6 till the end of the next century. Obviously, special FR breeder libraries were developed for this task in order to simulate properly fast growing regions. The impact of cross-sections uncertainties on MA mass flows and stocks was not assessed, but it was recognised that this topic should be of interest for successive WPFCs.

The same expert group studied the fuel cycle scenario codes benchmark (with codes COSI6, FAMILY-21, VISION, and DESAE2.2). In particular, 3 scenarios were analysed: UOX open-cycle, mono-recycling of plutonium in the PWRs, and mono-recycling of the plutonium in the PWRs and deployment of Gen-IV fast reactors. The main conclusions were that general trends provided by all the four considered codes were rather in good agreement especially for simplest cases (open cycle), although for most complex cases some discrepancies appeared due to differences in the way physical models are implemented and to the modelling capacity of the codes.

2.2.4 WPRS

The Working Party on Scientific Issues of Reactor Systems (WPRS) published a state-of-the-art report on Minor Actinide Burning in Thermal Reactors in 2013 [3]. Overall, the conclusion is that the burning of minor actinides in thermal reactors is technically feasible, certainly with respect to the impact on reactor neutronics parameters and might offer the potential to significantly reduce the quantities of some long-lived isotopes in spent fuel. However, various aspects of fuel management, particularly fabrication and handling are likely to present significant challenges, notably in the area of radiological protection. Also, it should be noted that the desired reduction of radiotoxicity in the resulting waste is inherently more limited in thermal reactors than in fast reactor systems.

To date, core design analyses have been carried out using the standard nuclear design codes validated for LEU and MOX cores. In LEU fuel, the inventories of the minor actinides are only of the order of 0.1% by mass and any nuclear data uncertainties for the minor actinides would only be expected to have a minor impact on in-core performance. For MOX fuels, the minor actinide inventories are considerably higher, but so far, there has been no indication that minor actinide cross-sections can affect the predictive power of the core design codes. However, it is always possible that unknown cancellation effects may be occurring that are masking any biases in minor actinide data and that the underlying biases may become evident at higher loadings.

Noting the possibility of uncertainties in predictive capabilities at higher loading, minor actinide nuclear data have already been identified as a priority area for nuclear data R&D. New cross-section evaluations for minor actinides will need to be assessed by sensitivity analyses to determine the effect on integral core parameters. The new evaluations will eventually need to be tested against integral experiments and against the first commercial-scale minor actinide irradiations to establish a full validation base for the nuclear data libraries and nuclear design codes. This can be expected to be a prolonged activity before the nuclear design methods can be considered mature for minor actinide applications.

2.2.5 IAEA MANREAD

The IAEA Co-ordinated Research Project (CRP) on Minor Actinide Nuclear Reaction Data (MANREAD) started in 2007. The aim of the project is to deliver a well-documented set of experimental data that form a reference database for evaluators of minor actinide. The MAs considered are: ^{234}U, ^{236}U, ^{237}Np, ^{238}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{242}m, ^{243}m, and $^{243\text{-}248}$Cm. The objectives of the CRP are:

- assessment of the available experimental data on the MAs for the total, capture, fission, inelastic scattering and (n,2n) cross-sections;
- a documented compilation of neutron cross-section data and their uncertainties for measurements performed by CRP participants on the MAs with a focus on information of relevance to evaluators;
- assessment of evaluated nuclear data for the MAs from the major libraries, including information on data uncertainties and covariances;
- assessment of achievable accuracies with contemporary measurement facilities and methodologies for the MAs;
- exploration of the potential of new experimental techniques and facilities for neutron cross-section measurements.

2.3 Important nuclear data

2.3.1 Evaluation of target accuracy of nuclear data

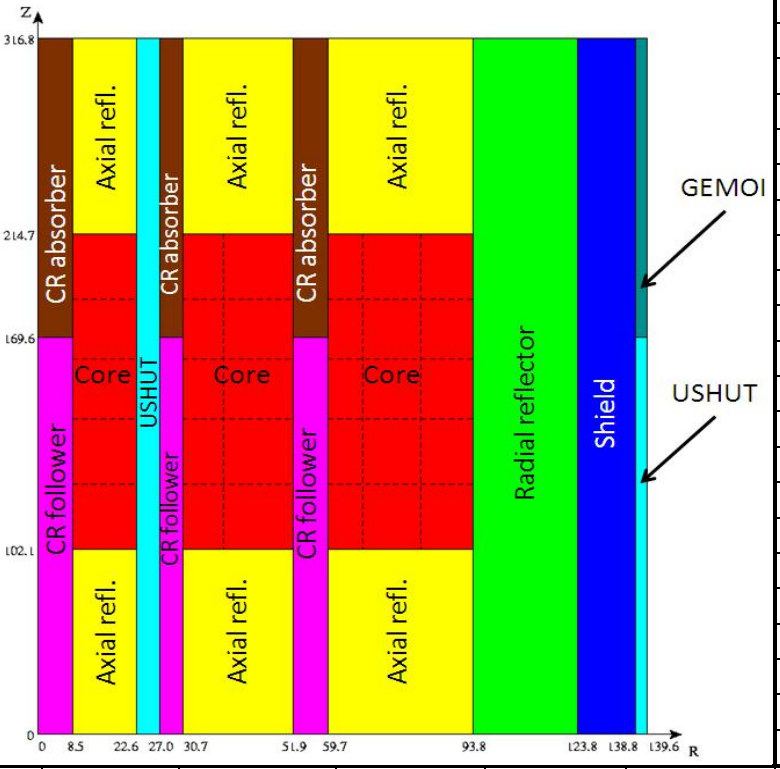
Innovative fuel cycles, in particular those that require the introduction of TRU burners, present several potential feasibility issues, especially in the field of development of new fuels with high-minor actinide (MA) content. Their definition, fabrication, performance under irradiation and reprocessing are crucial issues still under investigation. In this context, neutronics issues could seem to be of a lesser importance. However, key feasibility issues can arise in terms of nuclear data uncertainty impact, since safety and operation constraints can jeopardise specific fuel choices (e.g. maximum MA content) of potential interest.

Nuclear data requirements for innovative systems have been initially assessed, using sensitivity and uncertainty analysis and preliminary covariance data. Previous work was performed and preliminary indications were obtained [4]. The availability of more realistic uncertainty data allows a better target accuracy assessment. The results presented here differ from those of [4] in two respects: the method used is a more general one since it accounts for energy correlations in the definition of target accuracies [5] and it is based on a very recent release of new covariance data: COMMARA 2.0 [6].

The present analysis is related to two typical TRU burner systems [4]. The first system, called SFR, is a Na-cooled, metal-fuelled fast reactor loaded with TRU fuel of a composition as discharged by standard PWRs (i.e. Pu/MA ratio ~10). This is the typical composition of a TRU burner reactor when all TRUs are considered as potential waste. Figure 7 shows the details of the model and densities used for the SFR.

The second system, called ADMAB, is a Pb-cooled nitride-fuelled fast spectrum ADS; the fuel composition has a Pu/MA ratio close to one. This is a typical MA burner in a so-called double strata fuel cycle strategy where most of the Pu is considered as a resource and recycled in LWRs (or FRs) and MA are loaded in the burner ADS together with some Pu that is needed to create a fuel such that the reactivity loss during the cycle stays approximately close to zero (and consequently, the accelerator proton current stays ~constant). The details of the model and densities used for the ADMAB are reported in Figure 8.

Figure 7: SFR geometry and homogenised compositions [10^{24} at/cm^3], temperature: 850 K



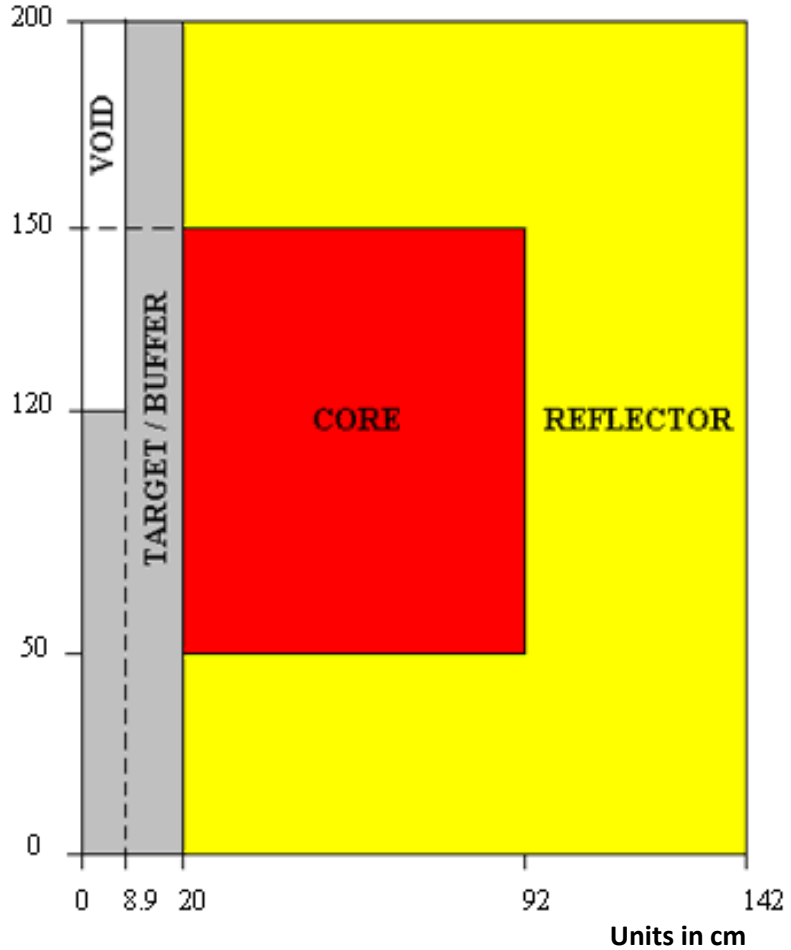
Average Fuel Compositions		
Isotope	Core	
U234	1.545E-5	
U235	5.030E-6	
U236	1.119E-5	
U238	1.697E-3	
Np237	8.626E-5	
Pu238	1.414E-4	
Pu239	7.325E-4	
Pu240	8.822E-4	
Pu241	1.610E-4	
Pu242	2.726E-4	
Am241	1.069E-4	
Am242m	7.408E-6	
Am243	9.517E-5	
Cm242	5.827E-6	
Cm243	5.688E-7	
Cm244	6.698E-5	
Cm245	1.738E-5	
Cm246	9.456E-6	
Fe	2.061E-2	
Cr	2.994E-3	
Ni	1.153E-4	
Mo	2.117E-4	
Zr	2.478E-3	
Na	1.099E-2	
Mn	2.218E-4	

Isotope	Axial Refl. (Bottom)	Axial Refl. (Top)	Radial Refl.	Shield	CR Absorber	CR Follower	USHUT	GEM0I
Fe	4.444E-2	1.909E-2	5.885E-2	1.789E-2	2.343E-2	2.343E-2	7.981E-3	7.981E-3
Cr	6.457E-3	2.774E-3	8.552E-3	2.599E-3	3.404E-3	3.404E-3	1.160E-3	1.160E-3
Ni	2.487E-4	1.068E-4	3.294E-4	1.001E-4	1.311E-4	1.311E-4	4.468E-5	4.468E-5
Mo	4.565E-4	1.961E-4	6.046E-4	1.837E-4	2.406E-4	2.406E-4	8.199E-5	8.199E-5
Na	8.120E-3	1.618E-2	3.635E-3	4.078E-3	6.493E-3	6.493E-3	1.968E-2	1.320E-3
Mn	4.783E-4	2.054E-4	6.335E-4	1.925E-4	2.521E-4	2.521E-4	8.591E-5	8.591E-5
B10				9.768E-3	6.326E-3			
B11				3.932E-2	2.546E-2			
C				1.227E-2	7.947E-3			

GEMOI: Gas Expansion Module, USHUT: Ultimate Shutdown

Figure 8: ADMAB geometry and homogenised compositions [10^{24} at/cm^3], temperature: 1 773 K

	Fuel		Reflector		Reflector		Reflector	
	Core							
Np237	4.377E-4		Fe54	2.990E-3	Ni58	1.977E-4	Bi	5.039E-3
Pu238	4.226E-5		Fe56	4.560E-2	Ni60	7.305E-5	W182	2.140E-5
Pu239	5.051E-4		Fe57	1.075E-3	Ni61	3.111E-6	W183	1.155E-5
Pu240	2.321E-4		Fe58	1.344E-4	Ni62	9.724E-6	W184	2.465E-5
Pu241	1.232E-4		Cr50	3.458E-4	Ni64	2.388E-6	W186	2.280E-5
Pu242	9.102E-5		Cr52	6.422E-3	Mo	3.565E-4	Target/Buffer	
Am241	8.084E-4		Cr53	7.134E-4	Mn	3.412E-4	Pb	1.320E-2
Am242m	1.089E-5		Cr54	1.741E-4	Pb	4.075E-3	Bi	1.632E-2
Am243	5.827E-4							
Cm242	4.079E-8							
Cm243	3.326E-6							
Cm244	2.371E-4							
Cm245	3.164E-5							
Cm246	5.355E-7							
Zr	7.477E-3							
N15	1.058E-2							
Fe54	9.759E-4							
Fe56	1.488E-2							
Fe57	3.507E-4							
Fe58	4.386E-5							
Cr50	1.128E-4							
Cr52	2.096E-3							
Cr53	2.328E-4							
Cr54	5.682E-5							
Ni58	6.451E-5							
Ni60	2.384E-5							
Ni61	1.015E-6							
Ni62	3.173E-6							
Ni64	7.792E-7							
Mo	1.163E-4							
Mn	1.114E-4							
Pb	6.360E-3							
Bi	7.865E-3							
W182	6.984E-6							
W183	3.770E-6							
W184	8.045E-6							
W186	7.439E-6							



The uncertainty, using COMMARA 2.0, on k_{eff} of both SFR and ADMAB systems is shown together with the major components (i.e. isotope, reaction) of the overall uncertainty in Section 2.3.b devoted to the uncertainty analysis. These uncertainties are much higher than the expected design requirement (±0.3 %△k/k). For both systems, the uncertainties have a significant impact. As an example, to account for the present uncertainty on the k_{eff} for the subcritical system ADMAB (~±1.7 % △k/k), it would be necessary to increase the nominal design value of the subcriticality level with a significant penalty on the proton beam power requirements.

In order to have a more compact way to assess the target accuracy, a 7-group energy structure has been adopted for this study. Table 3 shows the 7-energy group structure. The target accuracies, obtained with the method of Reference 2, on neutron cross-sections needed to meet the design requirements (in this particular case ±0.3 %△k/k of uncertainty for the k_{eff}), are given in Tables 4 and 5. The requirements are shown according (rank) to their overall original contribution to the total uncertainty on k_{eff}.

The requirements indicated in this study, and in particular for MA, are very small and probably beyond achievable limits with current experimental techniques for differential measurements. The results of the present investigation indicate that a careful analysis is still needed in order to define the most appropriate and effective strategy for data uncertainty reduction. Besides, a further consolidation of the present covariance data libraries, a strategy of combined use of integral and differential measurements should be pursued in order to meet future requirements, as proposed, e.g. in [7]. In particular, integral experiments devoted to improving the knowledge of nuclear data relevant to transmutation scenarios will be very useful for achieving this goal.

Table 3: Seven-energy groups structure (eV)

Group	Upper energy	Group	Upper energy
1	$1.96403\ 10^{7}$	5	$2.03468\ 10^{3}$
2	$2.23130\ 10^{6}$	6	$2.26033\ 10^{1}$
3	$4.97871\ 10^{5}$	7	$5.40000\ 10^{-1}$
4	$6.73795\ 10^{4}$		

Table 4: Nuclear data target accuracy requirements (first 20 major contributors) on a total uncertainty of 300 pcm on k_{eff} for the SFR system

Rank	Cross-section	Energy range (eV)	Current (%)	Required (%)
1	^{240}Pu ν Gr. 2	2.23x10^{6} to 4.98x10^{5}	3.2	0.9
2	^{245}Cm σ^{fis} Gr. 3	4.98x10^{5} to 6.74x10^{4}	47.0	3.2
3	^{56}Fe σ^{inel} Gr. 2	2.23x10^{6} to 4.98x10^{5}	11.4	1.8
4	^{56}Fe σ^{el} Gr. 3	4.98x10^{5} to 6.74x10^{4}	8.9	1.7
5	^{245}Cm σ^{fis} Gr. 2	2.23x10^{6} to 4.98x10^{5}	47.4	4.5
6	^{240}Pu ν Gr. 3	4.98x10^{5} to 6.74x10^{4}	4.8	1.5
7	^{240}Pu σ^{fis} Gr. 2	2.23x10^{6} to 4.98x10^{5}	2.5	1.1
8	^{238}U σ^{inel} Gr. 2	2.23x10^{6} to 4.98x10^{5}	16.7	2.8
9	^{245}Cm σ^{fis} Gr. 4	6.74x10^{4} to 2.03x10^{3}	15.9	3.7
10	^{56}Fe σ^{cap} Gr. 3	4.98x10^{5} to 6.74x10^{4}	12.2	2.7
11	^{56}Fe σ^{el} Gr. 2	2.23x10^{6} to 4.98x10^{5}	5.8	2.1
12	^{240}Pu ν Gr. 1	1.96x10^{7} to 2.23x10^{6}	2.5	1.5
13	^{238}U σ^{inel} Gr. 1	1.96x10^{7} to 2.23x10^{6}	19.4	3.7
14	^{240}Pu σ^{fis} Gr. 3	4.98x10^{5} to 6.74x10^{4}	6.2	2.0
15	^{240}Pu σ^{cap} Gr. 2	2.23 10^{6} to 4.98 10^{5}	33.4	4.1
16	^{242}Pu σ^{cap} Gr. 4	6.74x10^{4} to 2.03x10^{3}	20.2	3.9
17	^{244}Cm σ^{fis} Gr. 2	2.23x10^{6} to 4.98x10^{5}	14.4	3.2
18	^{242}Pu σ^{cap} Gr. 3	4.98x10^{5} to 6.74x10^{4}	28.2	4.6
19	^{240}Pu σ^{cap} Gr. 4	6.74x10^{4} to 2.03x10^{3}	4.5	2.1
20	^{240}Pu ν Gr. 4	6.74x10^{4} to 2.03x10^{3}	4.6	2.2

Table 5: Nuclear data target accuracy requirements (first 20 major contributors) on a total uncertainty of 300 pcm on k_{eff} for the ADMAB system

Rank	Cross-section	Energy range (eV)	Current (%)	Required (%)
1	^{245}Cm σ^{fis} Gr. 3	4.98x10^{5} to 6.74x10^{4}	47.0	1.5
2	^{244}Cm σ^{fis} Gr. 2	2.23x10^{6} to 4.98x10^{5}	14.3	1.0
3	^{245}Cm σ^{fis} Gr. 2	2.23x10^{6} to 4.98x10^{5}	47.5	2.0
4	^{244}Cm ν Gr. 2	2.23x10^{6} to 4.98x10^{5}	8.7	0.9
5	^{244}Cm σ^{cap} Gr. 4	6.74x10^{4} to 2.03x10^{3}	64.4	2.5
6	^{243}Am σ^{fis} Gr. 2	2.23x10^{6} to 4.98x10^{5}	8.8	1.0
7	^{245}Cm σ^{fis} Gr. 4	6.74x10^{4} to 2.03x10^{3}	14.7	1.6
8	^{244}Cm σ^{cap} Gr. 3	4.98x10^{5} to 6.74x10^{4}	48.6	2.7
9	^{244}Cm σ^{fis} Gr. 1	1.96x10^{7} to 2.23x10^{6}	19.7	1.8
10	^{243}Am σ^{inel} Gr. 2	2.23x10^{6} to 4.98x10^{5}	21.6	2.0
11	^{243}Am σ^{cap} Gr. 2	2.23x10^{6} to 4.98x10^{5}	6.8	1.2
12	^{243}Am σ^{fis} Gr. 1	1.96x10^{7} to 2.23x10^{6}	12.1	1.6
13	^{243}Am σ^{cap} Gr. 3	4.98x10^{5} to 6.74x10^{4}	8.2	1.3
14	^{244}Cm σ^{cap} Gr. 2	2.23x10^{6} to 4.98x10^{5}	77.9	4.3
15	^{244}Cm ν Gr. 1	1.96x10^{7} to 2.23x10^{6}	10.1	1.7
16	^{244}Cm σ^{fis} Gr. 3	4.98x10^{5} to 6.74x10^{4}	10.0	2.0
17	^{209}Bi σ^{inel} Gr. 2	2.23x10^{6} to 4.98x10^{5}	14.0	1.9
18	^{15}N σ^{elas} Gr. 2	2.23x10^{6} to 4.98x10^{5}	2.9	0.9
19	^{245}Cm σ^{fis} Gr. 1	1.96x10^{7} to 2.23x10^{6}	38.0	4.1
20	^{244}Cm ν Gr. 3	4.98x10^{5} to 6.74x10^{4}	10.0	1.9

2.3.2 Comparison of uncertainty analysis among nuclear data libraries

In the previous section, evaluation results for target accuracy of nuclear data based on the study performed by WPEC Subgroup 26 were described. The requirements lists cover the energy range of interest and the range of uncertainty reduction with respect to the BOLNA covariance data for SFR and ADMAB system. However, since the uncertainty prediction required for the evaluation of target accuracy depends on the covariance data, it is possible to provide different priority list in case using other covariance data. While the discussions of the covariance data is beyond the scope of this expert group, the benchmark uncertainty analyses were performed in order to discuss the necessary integral experiments considering the requirements based on the uncertainty analysis results by various covariance data.

The benchmark uncertainty analysis was carried out for two typical TRU burner systems described in the previous section. The first system, called SFR, is a Na-cooled, metal-fuelled fast reactor loaded with TRU fuel of a composition as discharged by standard PWRs (i.e. Pu/MA ratio ~10). The second system, called ADMAB, is a Pb-cooled nitride-fuelled fast spectrum ADS; the fuel composition has a Pu/MA ratio close to one. This is a typical MA burner in a so-called double strata fuel cycle strategy where most of the Pu is considered as a resource and recycled in LWRs (or FRs) and MA are loaded in the burner ADS together with some Pu. The details of the model and densities used for both SFR and ADMAB are reported in Figures 7 and 8 in Section 2.3.1. The benchmark analysis focused on the uncertainty of k_{eff} values due to the neutron cross-section uncertainties.

The uncertainty analysis results with the cross-section covariance matrix from COMMARA-2.0, JENDL-4.0 [8], and JEF-2.2 [9] with ERALIB1 [10] were provided by the Expert Group members.

The COMMARA-2.0 covariance library was developed by BNL-LANL collaboration for the Advanced Fuel Cycle Initiative applications over the period of three years, 2008-2010. It contains covariances for 110 materials relevant to fast reactor R&D. The library should be used together with the ENDF/B-VII.0 central values of the latest official release of the US files of the evaluated neutron cross-sections. COMMARA-2.0 library contains neutron cross-section covariances for 12 light nuclei (coolants and moderators), 78 structural materials and fission products, and 20 actinides. Covariances are given in 33-energy groups, from 10–5 eV to 19.6 MeV, obtained by processing with LANL processing code NJOY using 1/E flux. In addition to these 110 files, the library contains 20 files with nu-bar covariances, 3 files with covariances of prompt fission neutron spectra (^{238,239,240}Pu), and 2 files with mu-bar covariances (^{23}Na, ^{56}Fe). For the covariances of fast region fission cross-section of major actinides, fission is obtained from detailed analysis of experimental data, while other reaction channels are evaluated mostly by nuclear reaction model code GNASH with experimental data included through the Bayesian code KALMAN [11]. In the resonance region, evaluations are based either on a full-scale SAMMY analysis or retroactive SAMMY analysis.

Covariance data of actinide nuclear data stored in JENDL-4.0 have been evaluated based on available experimental data and reaction model calculation. The covariance matrix was basically deduced applying the consistent method with the nuclear data evaluation. For the fast neutron fission cross-sections of 6 major actinides of ^{233,235,238}U and ^{239,240,241}Pu were evaluated simultaneously with experimental data of both cross-sections and their ratios using the least-squares fitting, resulting in the covariance matrices that have correlations between the data of different nuclei. For the minor actinide, the least-squares fitting was used for fission cross-section evaluation and the covariances were obtained from the same calculation. For other reaction cross-sections, covariances were evaluated using KALMAN code with the sensitivities calculated by CCONE code [12]. Covariances for other data such as resonance parameters and average numbers of fission neutrons were also evaluated. The JENDL-4.0 library contains covariance data of 1) average number of neutrons per fission, 2) resonance parameters, 3) cross-sections,

4) angular distributions of elastic scattering, and 5) prompt fission spectra for all 79 actinides.

The nuclear data library ERALIB1 has been obtained from the JEF-2.2 library by a consistent statistical adjustment based on a generalised least-squares method using over 350 integral data from 71 different systems.

In the analyses with BOLNA and JEF, the sensitivity calculations were performed with the ERANOS code system [13], which allows calculating homogeneous and inhomogeneous solutions of the Boltzman equations. The discrete ordinate module BISTRO [14] was used to perform flux and generalised importance function calculation. An S4P1 approximation in RZ geometry proved to be sufficiently accurate for this type of calculation. Cross-section data were processed with ECCO code [15].

In the analysis with JENDL, the SAGEP code [16], based on the generalised perturbation theory and diffusion theory, was used to calculate the cross-section sensitivity coefficient with JENDL-4.0 library. The effective cross-sections were calculated by SLAROM [17] in 70 groups. The 70-group cross-sections are collapsed to 18 groups, and the collapsed cross-sections were used for the uncertainty analysis. For the uncertainty analysis, the cross-section adjustment code ABLE [18] was used. The ABLE predicts the uncertainty of neutronics integral parameters using the sensitivity coefficient obtained by SAGEP. In order to utilise the covariance data in JENDL, a covariance processing code ERRORJ [19] was used to calculate covariance of multi-group cross-section.

The calculation results for k_{eff} and the corresponding uncertainty expressed in pcm are shown in Table 6. The general tendency for the calculating results of k_{eff} is that the values by ENDF/B-VII and JEND-4.0 provide similar values, and they are about 2-3 % larger than the values by JEF-2.2 and ERALIB1. For the uncertainty results, the calculation results for ADMAB are larger than those for SFR. It will be deduced from the difference of MA content for both transmutation systems. The results are almost the same for COMMARA-2.0 and JENDL-4.0 results, while the results by JENDL-4.0 are slightly smaller (about 200 pcm) than those by COMMARA-2.0. On the other hand, while the results by JEF-2.2 provide larger values than other results, these values are reduced by using the ERALIB1 covariance data. The uncertainty reduction is drastic in the case of SFR. The comparison with the results by JEF-2.2 and ERALIB1 clearly shows that the cross-section adjustment is effective for reduction of the calculation uncertainty, especially when enough integral experiments are available.

Table 6: Calculation results for k_{eff} and uncertainty in pcm of SFR and ADMAB

	SFR		ADMAB	
Library	k_{eff}	Uncertainty [pcm]	k_{eff}	Uncertainty [pcm]
ENDF\B-VII + COMMARA-2.0	1.04745	1014[1]	0.97468	1668[1)]
JENDL-4.0	1.04585	786	0.97973	1403
JEF-2.2	1.02278	1630	0.94516	1986
(ERALIB1)	1.03389	659	0.95239	1616

1. The results by COMMARA-2.0 are the sub total of the first ten major isotope contributors.

Table 7 and 8 provide the detailed uncertainty breakdown by isotopes and reaction for k_{eff} of SFR and ADMAB, respectively. Figures 9 and 10 show the uncertainty breakdown for actinide nuclides. It should be noted that the results by COMMARA-2.0 are the subtotal of the first ten major isotope contributors. Comparing the results, it can be confirmed that there are significant differences for each contributor. For example, for ^{245}Cm, the fission reaction is the main contributor in the COMMARA-2.0 and RALIB-1 calculations, but this is not so large in the JENDL-4.0 and JEF-2.2 calculations. From these results, while it is difficult to identify the major sources for the uncertainty, it is possible to indicate the important nuclides and reactions to reduce the uncertainty for MA transmutation system. The important nuclides and reactions are listed below:

Main actinides (U and Pu):

σ_{cap} : ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{242}Pu

σ_{fis} : ^{238}Pu, ^{239}Pu, ^{240}Pu

ν : ^{238}Pu, ^{240}Pu

σ_{inel} : ^{238}U

Minor actinides (Np, Am, and Cm):

σ_{cap} : ^{237}Np, ^{241}Am, ^{243}Am, ^{244}Cm

σ_{fis} : ^{237}Np, ^{241}Am, ^{242m}Am, ^{243}Am, ^{244}Cm, ^{245}Cm

ν : ^{241}Am, ^{243}Am, ^{244}Cm, ^{245}Cm

σ_{inel} : ^{237}Np, ^{241}Am, ^{243}Am

The uncertainty results show that these values are much higher than the expected design requirement (300 pcm). To achieve this target requirement, the uncertainties for the main contributor should be reduced. The comparison between the results by JEF-2.2 and ERALIB1 is instructive. The uncertainties for Pu isotopes are effectively reduced by the cross-section adjustment method using adequate integral experiments.

Table 7: Uncertainty analysis results for k_{eff} of SFR

(a) BOLNA-2.0

Isotope	σ_{cap}	σ_{fiss}	ν	σ_{el}	σ_{inel}	Total
^{238}U	50	9	33	51	230	244
^{238}Pu	117	119	154	8	32	229
^{239}Pu	131	136	41	27	78	211
^{240}Pu	206	241	439	39	113	554
^{241}Pu	95	45	34	2	38	117
^{242}Pu	176	93	81	7	15	216
^{243}Am	55	58	13	2	18	83
^{244}Cm	147	141	111	2	5	232
^{245}Cm	8	444	50	0	3	447
^{56}Fe	195	0	0	320	321	493
Total	423	567	492	328	422	1014

(b) JENDL-4.0

Isotope	σ_{cap}	σ_{fiss}	ν	σ_{el}	σ_{inel}	Total
^{235}U	1	12	1	0	0	12
^{238}U	110	34	18	17	88	147
^{237}Np	49	12	8	1	4	51
^{238}Pu	177	93	161	3	6	256
^{239}Pu	119	205	80	14	21	251
^{240}Pu	164	99	29	12	25	195
^{241}Pu	51	117	74	7	14	148
^{242}Pu	111	43	27	13	12	123
^{241}Am	94	13	16	2	4	96
^{242m}Am	14	18	10	0	0	25
^{243}Am	120	16	11	4	12	122
^{242}Cm	9	10	9	0	0	16
^{243}Cm	1	2	2	0	0	3
^{244}Cm	78	31	92	1	3	125
^{245}Cm	33	45	66	0	2	86
^{246}Cm	7	4	4		1	9
^{23}Na	10	0	0	10	160	160
^{52}Cr	4	0	0	31	12	33
^{56}Fe	152	0	0	292	455	562
^{58}Ni	3	0	0	1	1	3
Total	394	285	230	295	492	786

(c) JEF-2.2

Isotope	σ_{cap}	σ_{fiss}	ν	σ_{el}	σ_{inel}	Total
^{237}Np	0	69	0	0	0	69
^{238}Pu	47	433	108	2	10	449
^{239}Pu	119	899	326	7	35	964
^{240}Pu	334	788	263	18	54	897
^{241}Pu	97	530	87	3	8	546
^{242}Pu	84	175	58	6	15	203
^{241}Am	129	68	33	2	10	149
^{242m}Am	2	90	15	0	1	92
^{243}Am	99	43	22	2	27	113
^{242}Cm	2	9	3	0	0	10
^{243}Cm	0	8	3	0	0	8
^{244}Cm	24	72	29	1	7	81
^{245}Cm	9	260	47	0	2	264
Zr	22	0	0	32	96	104
^{52}Cr	17	0	0	134	29	139
^{56}Fe	92	0	0	263	385	476
^{58}Ni	3	0	0	2	2	4
^{10}B	46	0	0	16	0	49
Total	427	1421	450	299	405	1630

(d) ERALIB

Isotope	σ_{cap}	σ_{fiss}	ν	σ_{el}	σ_{inel}	Total
^{237}Np	0	72	0	0	0	72
^{238}Pu	45	429	110	3	11	445
^{239}Pu	34i	127	6i	2i	5i	122
^{240}Pu	38	126	166	12i	55	218
^{241}Pu	11	130	63	2i	9	146
^{242}Pu	21	44	55	6	15	75
^{241}Am	124	70	34	2	11	147
^{242m}Am	2	88	15	0	2	89
^{243}Am	95	45	23	2	30	111
^{242}Cm	2	9	3	0	1	10
^{243}Cm	0	87	3	0	0	8
^{244}Cm	23	74	30	1	7	84
^{245}Cm	8	253	47	0	2	258
Zr	22	0	0	36	98	107
^{52}Cr	12	0	0	36i	25	22i
^{56}Fe	60	0	0	45	80	110
^{58}Ni	5i	0	0	8i	2i	9i
^{10}B	44	0	0	18	0	48
Total	185	570	227	47	145	659

Table 8: Uncertainty analysis results for k_{eff} of ADMAB

(a) BOLNA-2.0

Isotope	σ_{cap}	σ_{fiss}	ν	σ_{el}	σ_{inel}	Total
^{237}Np	166	162	63	2	102	261
^{239}Pu	89	122	34	13	87	178
^{240}Pu	52	91	150	9	73	197
^{241}Am	207	125	173	11	120	321
^{242m}Am	11	133	10	0	5	134
^{243}Am	316	430	92	3	266	604
^{244}Cm	555	642	505	1	60	989
^{245}Cm	12	1040	111	0	10	1046
^{15}N	1	0	0	214	12	215
^{209}Bi	54	0	0	64	180	198
Total	702	1327	577	224	381	1668

(b) JENDL-4.0

Isotope	σ_{cap}	σ_{fiss}	ν	σ_{el}	σ_{inel}	Total
^{237}Np	227	83	53	1	43	251
^{238}Pu	47	39	62	0	4	87
^{239}Pu	81	189	70	2	32	220
^{240}Pu	42	48	10	1	15	66
^{241}Pu	37	117	70	1	22	143
^{242}Pu	39	19	12	1	8	46
^{241}Am	601	128	165	3	57	639
^{242m}Am	18	37	17	0	3	44
^{243}Am	664	132	82	5	162	701
^{242}Cm	0	0	0	0	0	0
^{243}Cm	5	15	11	0	1	19
^{244}Cm	262	149	434	1	26	529
^{245}Cm	50	111	151	0	7	195
^{15}N	0	0	0	255	0	255
^{52}Cr	3	0	0	10	17	20
^{56}Fe	79	0	0	117	659	674
^{58}Ni	1	0	0	0	1	2
^{209}Bi	0	0	0	182	0	182
Total	972	362	512	335	684	1385

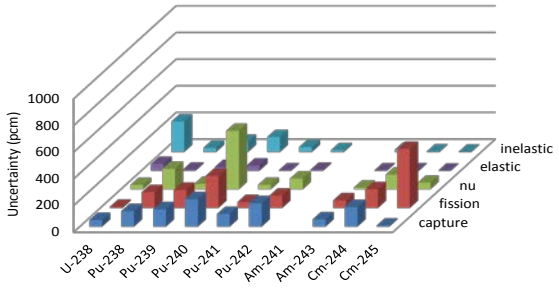
(c) JEF-2.2

Isotope	σ_{cap}	σ_{fiss}	ν	σ_{el}	σ_{inel}	Total
^{237}Np	0	435	0	0	0	435
^{238}Pu	11	162	37	0	6	167
^{239}Pu	68	791	259	1	50	837
^{240}Pu	76	261	82	1	28	286
^{241}Pu	67	506	77	0	12	516
^{242}Pu	25	74	23	0	9	82
^{241}Am	803	626	280	2	151	1067
^{242m}Am	3	171	27	0	4	173
^{243}Am	496	323	153	3	312	686
^{242}Cm	0	0	0	0	0	0
^{243}Cm	1	55	18	0	0	58
^{244}Cm	65	312	118	1	46	344
^{245}Cm	13	575	100	0	6	584
Zr	60	0	0	41	498	503
^{52}Cr	9	0	0	10	34	37
^{56}Fe	47	0	0	59	558	563
^{58}Ni	2	0	0	0	2	3
Total	958	1459	456	73	828	1986

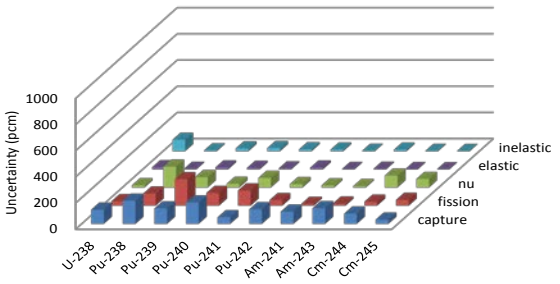
(d) ERALIB

Isotope	σ_{cap}	σ_{fiss}	ν	σ_{el}	σ_{inel}	Total
^{237}Np	0	438	0	0	0	438
^{238}Pu	11	162	37	0	6	166
^{239}Pu	41i	145	59	2i	35	155
^{240}Pu	28i	38	33	3i	26	49
^{241}Pu	11	133	64	1i	12	149
^{242}Pu	8i	16	21	0	9	27
^{241}Am	793	630	280	2	155	1063
^{242m}Am	3	169	27	0	4	171
^{243}Am	489	325	153	3	325	688
^{242}Cm	0	0	0	0	0	0
^{243}Cm	1	55	18	0	1	57
^{244}Cm	64	316	120	1	47	347
^{245}Cm	13	569	99	0	6	578
Zr	62	0	0	32	482	487
^{52}Cr	5	0	0	11	28	31
^{56}Fe	32	0	0	30i	110	111
^{58}Ni	3i	0	0	1	1i	3i
Total	936	1103	370	616	616	1616

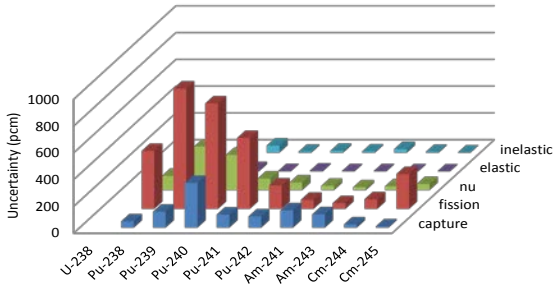
Figure 9: Uncertainty analysis results for k_{eff} of SFR (actinide isotopes)



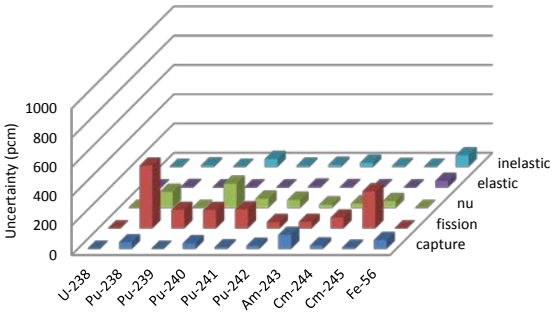
COMMARA-2.0



JENDL-4.0

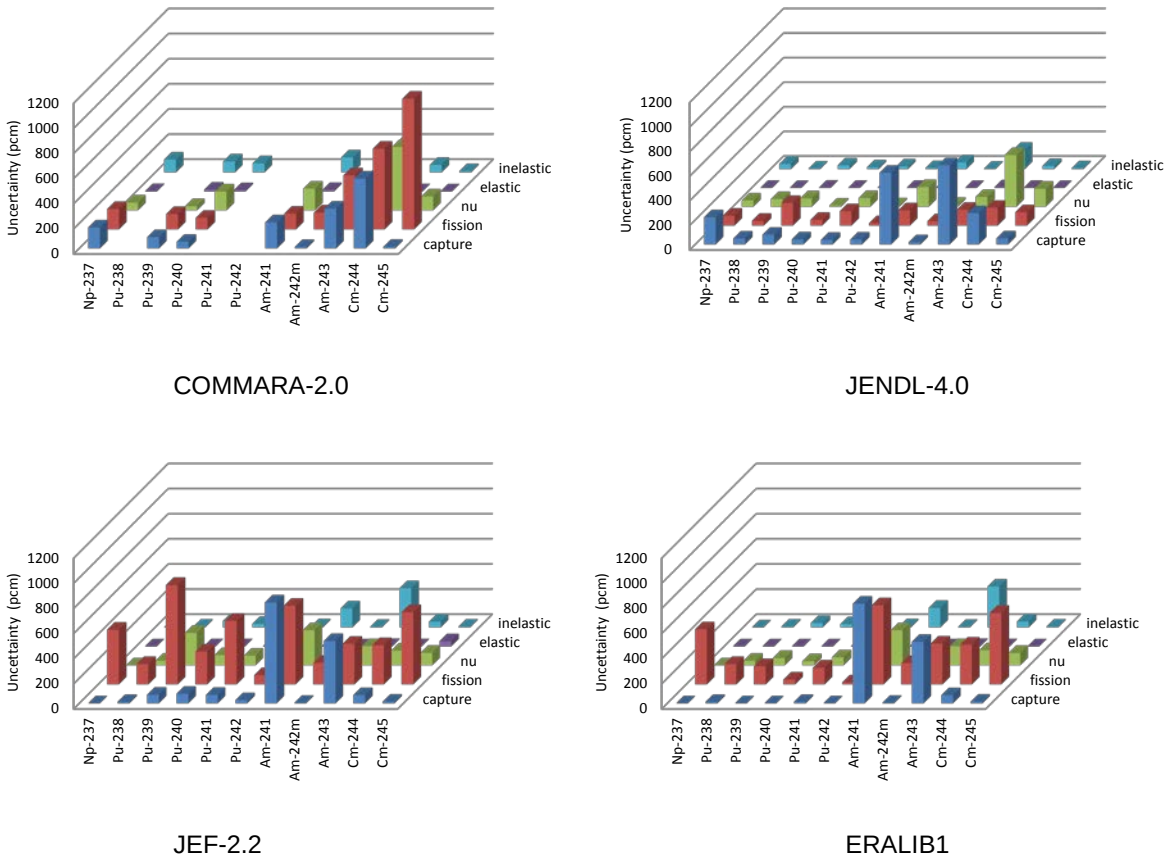


JEF-2.2



ERALIB1

Figure 10: Uncertainty analysis results for k_{eff} of SFR (actinide isotopes)



2.3.3 Identification of important nuclear data in fuel cycle

2.3.3.1 General aspects and calculations assumptions

In the present chapter, the relative importance of the different heavy nuclides (in particular plutonium isotopes and minor actinides) has been assessed in closed fuel cycles where critical fast reactors with various conversion ratios are implemented. For this purpose, a regional-type scenario was chosen (Figure 11), where two groups of nations are considered:

- Group A: Phasing-out group of nations, with legacy spent fuel inventory represented in COSI6 simulations as a decaying stock [20].
- Group B: On-going nuclear policy group of nations, operating in particular a UOX-PWR fleet. Target fuel burn-up of 50 GWd/t_{HM} and an irradiated fuel cooling time of 5 years are adopted.
- A regional group of facilities, shared by the two groups of nations, which has the final goal to transmute till the end of the present century all TRU from Group A and to stabilise the MA or TRU production (according to the specific objective) of Group B.

Figure 11: Scenario flow scheme

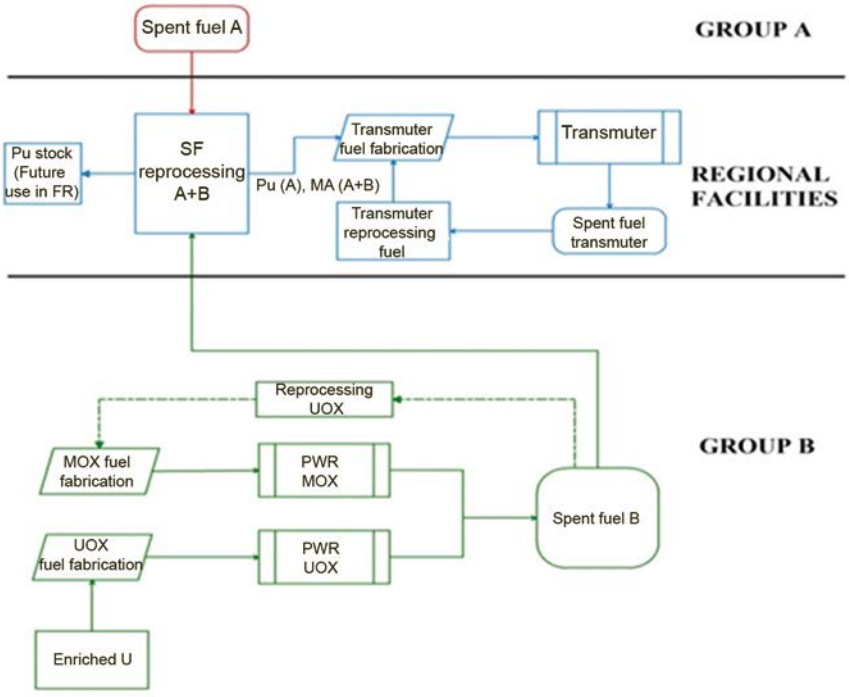
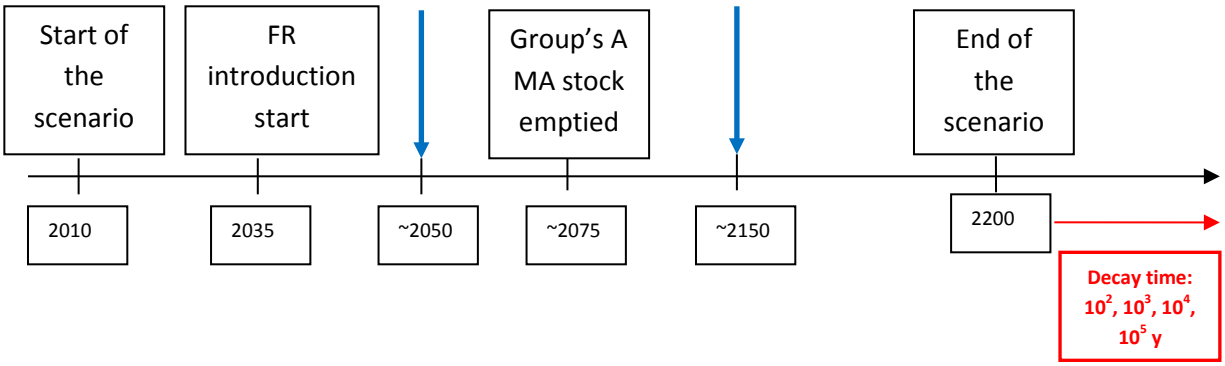


Figure 12 shows the most relevant dates and events of the scenario simulation.

In the present work, three types of transmuters, all fast critical reactors but with three different conversion ratios (CR), and two different minor actinides to plutonium ratios (MA/Pu) were investigated in order to explore a wide range of fuel cycle options. In particular:

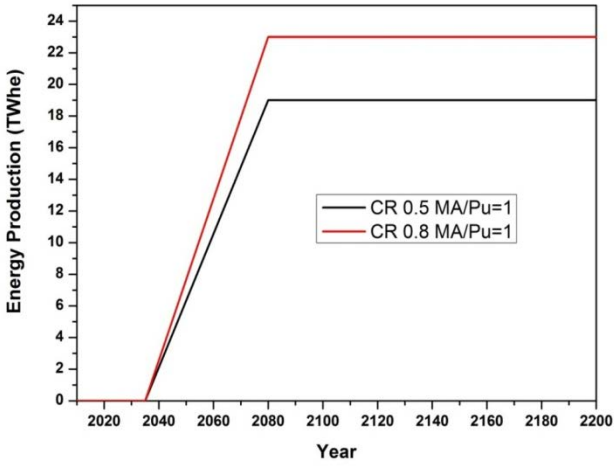
- Option 1: break-even FR with CR=1, MA/Pu=0.1.
- Option 2: burner FR with CR=0.8:
  - MA/Pu=0.1;
  - MA/Pu=1.
- Option 3: burner FR with CR=0.5:
  - MA/Pu=0.1;
  - MA/Pu=1.

Figure 12: Relevant dates in the scenario (blue arrows refer to fuel compositions evaluation dates)



With options 2 and 3, MA are produced by a PWR-UOX fleet whose energy production is kept constant (400 TWh_e) over the whole scenario simulation time (Figure 13), and FR transmuters are introduced as an extra dedicated component of the power park.

Figure 13: FRs with MA/Pu=1 fleet energy production



In the case of option 1, a replacement of the PWR fleet with a corresponding (i.e. equal total power) FR fleet has been simulated. In 2035, PWRs will start to be shut down and replaced by FRs up to 2080, and the total energy production is maintained constant (400 TWh_e).

The adoption of two different MA/Pu ratios in fresh FR fuel and of different values of the conversion ratio CR, allows fulfilling different objectives. In summary, if the full TRU inventory of countries "A" has to be eliminated by the end of this century and the MA inventory of countries "B" has to be stabilised, while the Pu inventory is kept growing (e.g. for future use in FRs in countries B), the dedicated regional burners (with CR=0.5 or CR=0.8) should have a fuel with a MA/TRU ratio equal ~1. The number of dedicated burners is consequently relatively small, as it is illustrated in Figure 13, where the fleet of dedicated regional burners is shown for both CR=0.5 and 0.8 in terms of energy production. In this case, the MA inventory is stabilised and the Pu inventory is growing, as required (Figures 14 and 15). It should be noted that in this fuel cycle option the reprocessing of the fuel from countries "A" and "B" requires the separation of Pu from MA.

Figure 14: Pu mass in complete fuel cycle for the CR<0 (fuel cycle option 1) and MA/Pu=1: the plutonium inventory is still growing

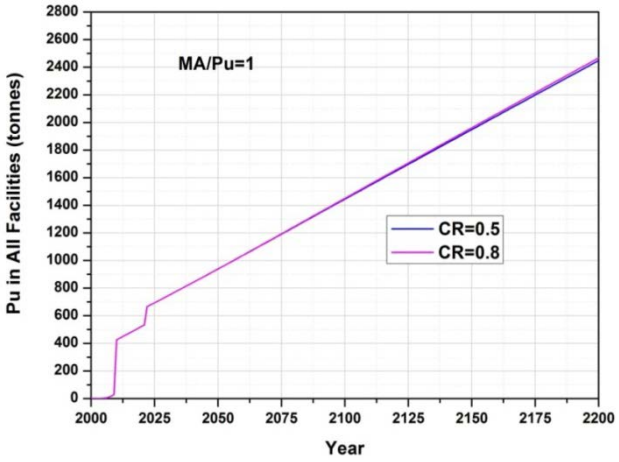
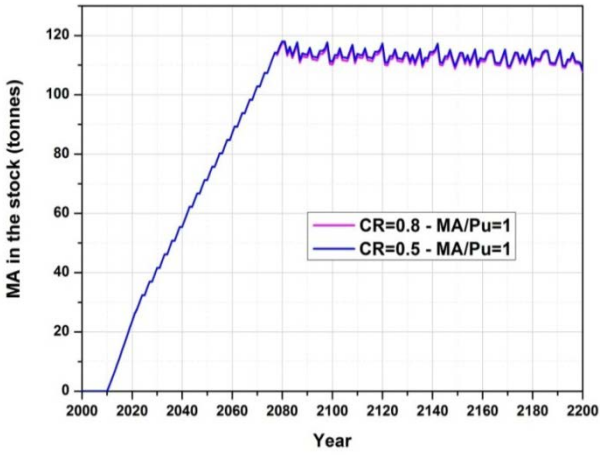


Figure 15: MA mass in the stock for CR<1 and MA/Pu=1 core (fuel cycle option 1)



If the full TRU inventory of countries "A" has to be eliminated by the end of this century and both MA and Pu inventories of group "B" should be stabilised (i.e. the Pu inventory of countries B has also to be reduced), the dedicated regional burners with CR=0.5 and CR=0.8 have a fuel with the MA/Pu ratio equal to 0.1, i.e. close to the ratio of MA/Pu in the fuel unloaded from the PWRs. It is not necessary to separate Pu from MA in the reprocessing step if that ratio is adopted. In this case, the number of dedicated transmuters is much higher than in the previous case since the burning requirements are much higher (since they concern both MA and Pu). Figure 16 gives the size of the burner fleet in terms of energy production. This fleet is very large in particular in the case of CR=0.8, since in this case the amount of MA that can be burnt is relatively modest. Figures 17 and 18 show the Pu and MA inventory stabilisation; for CR=0.8 – MA/Pu=0.1 core case, a proper stabilisation is not achievable due to the large amount of TRUs generated by the PWRs-UOX fleet compared with the transmuter transmutation capability (the TRUs stabilisation target is possible for lower PWRs energy production).

Figure 16: PWRs and FRs fleet energy production (total values are not shown)

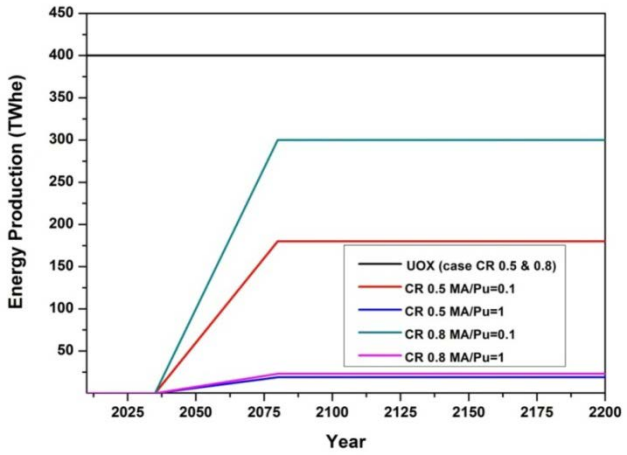


Figure 17: Pu mass in complete fuel cycle for the CR<0 and MA/Pu=0.1 (fuel cycle option 2): the plutonium inventory is roughly stabilised

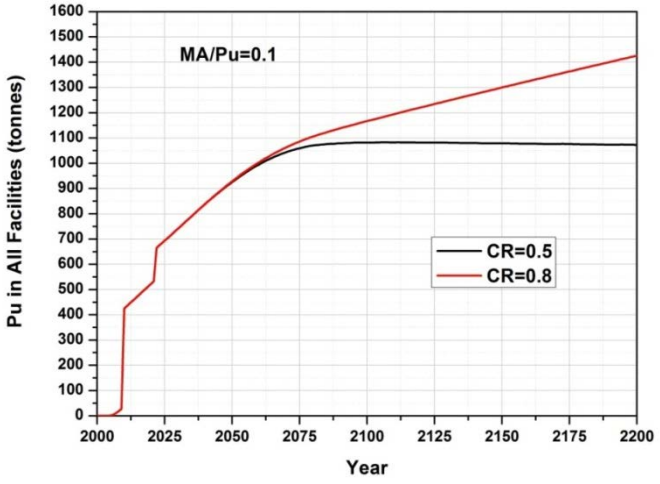
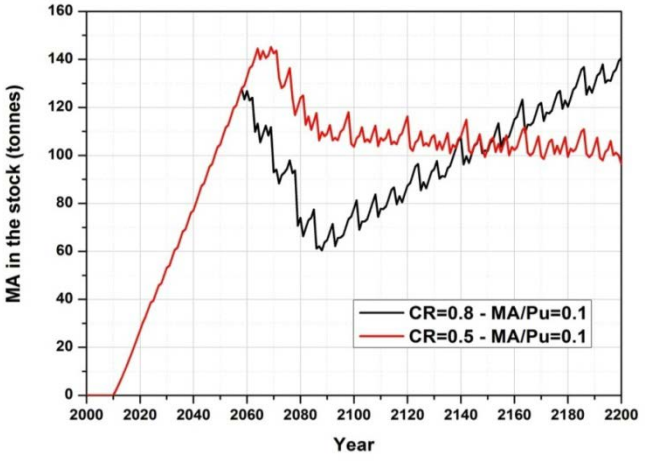


Figure 18: MA mass in the stock for CR<1 and MA/Pu=0.1 core (fuel cycle option 2)



a. The case with FRs with CR=1 (Figure 19) is quite different in purpose. In the previous cases, the PWR-UOX fleet stays constant and an extra dedicated fleet of burner FRs is deployed, while in this case, the PWRs fleet is replaced by a corresponding fleet of FRs that will manage the full TRU stock of both groups of countries. For this reason, a direct comparison of the cases a) b) and c) is possible only in relative terms (e.g. comparing the specific decay heat of unloaded fuels in each case). The Pu and MA inventories are given, respectively in Figures 20 and 21. The stabilisation of the MA inventory at a lower value (~60 t) is due to the fact that a smaller overall fleet of reactors is considered.

Figure 19: PWRs and FRs fleet energy production for CR=1 – MA/Pu=0.1 case (fuel cycle option 3)

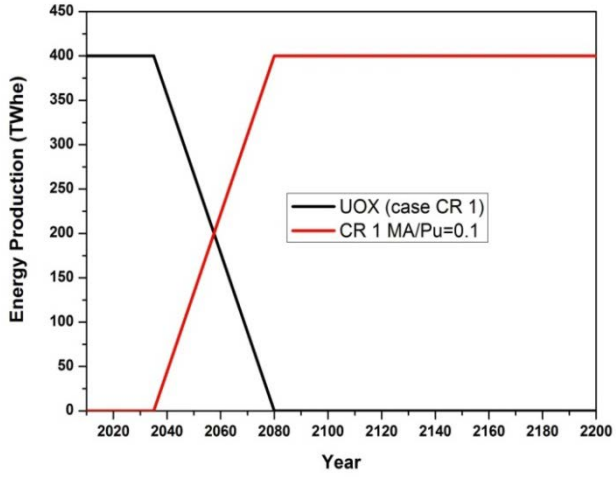


Figure 20: Pu mass in complete fuel cycle for an FR fleet with CR=1 and MA/Pu=0.1 replacing the UOX-PWR fleet: the plutonium inventory is roughly stabilised

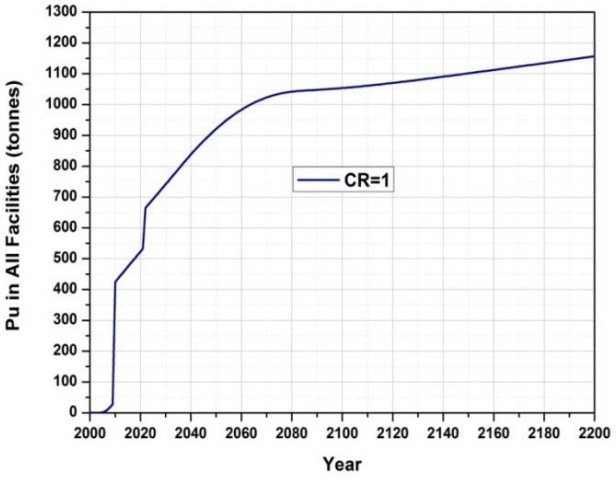
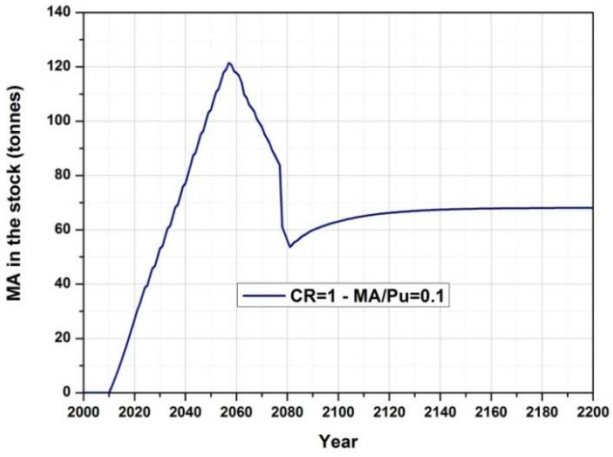


Figure 21: MA mass in the stock for CR=1 and MA/Pu=0.1 core (fuel cycle option 3)



There are two types of stocks which are transmuted (i.e. from Group A and Group B – see above), with different compositions, which strongly evolve over time, therefore, the composition in the fabrication plant was extracted around the half of the present century (when Group's A TRU stock is transmuted) and around the half of the next century (when Group's B MA inventory-is transmuted and stabilised).

In order to reduce calculations complexity, it was assumed that reprocessing compositions would be very similar to those of fabrication plants; assumed fabrication and reprocessing times are rather short (i.e. 0.5 years), while fuel cooling time is 5 years for PWRs and 2 years for FRs.

The fuel composition was extracted from the repository too, and the results were assessed at τ = 0, 10^3, 10^4, 10^5 years, and decay calculations were performed with the decay engine of NUCLEONICA (library JEFF-3.1 – Feb.2005, ICRP 72); the following four quantities were assessed:

- SF rate;
- ingestion radiotoxicity;
- isotopic power ($\alpha+\beta+\gamma$) – i.e. decay heat;
- γ dose rate.

The threshold of 10% relative contribution was applied to simplify the analysis of the results. This means that the isotope contributions to the four quantities quoted above lower than 10% were neglected. In order to illustrate the effect of this simplification in the presentation of the results, Table 9 shows a decay heat table where a smaller threshold value of 2% has been used.

The contribution of ^{242m}Am and ^{242}Am to dose rate (often comparable to that of ^{241}Am in the results) is caused by the default threshold of 15 keV in NUCLEONICA (below this energy level gamma rays are not able to penetrate skin's outer layers, so they are neglected); gamma dose rate (such as spontaneous fission rate) was evaluated in air at 1 meter only in the fabrication plant.

Only reprocessing losses of U and TRU (separation efficiency fixed to 99.9%) are stored in the repository and FP have not been considered. FR reactors are introduced in the cycle between 2035 and 2080.

Table 9: Decay heat table with 2% threshold

Case:		CR=0.5		CR=0.8		CR=1
Decay heat (α+β+γ)		MA/Pu=0.1	MA/Pu=1	MA/Pu=0.1	MA/Pu=1	MA/Pu=0.1
T=30 d	Tot (W/g)	0.35	0.41	0.17	0.29	0.87
Reactor discharge	^{238}Pu	7.0 %	8.9 %	7.6 %	9.0 %	5.4 %
2050	^{242}Cm	83.8 %	76.4 %	84.3 %	76.8 %	88.0 %
	^{244}Cm	7.5 %	12.5 %	5.9 %	11.8 %	4.5 %
T=30 d	Tot (W/g)	0.37	0.35	0.12	0.23	0.17
	^{238}Pu	7.4 %	16.9 %	7.3 %	17.7 %	6.7 %
Reactor discharge	^{242}Cm	79.4 %	68.7 %	68.5 %	67.7 %	75.3 %
2150	^{244}Cm	11.4 %	12.3 %	22.0 %	12.4 %	14.3 %
T=0.5 y	Tot (W/g)	0.023	0.047	0.021	0.033	0.09
Fabrication plants	^{238}Pu	36.9 %	24 %	40.6 %	26.1 %	40.3%
	^{239}Pu	2.0 %	-	-	-	2.2 %
2050	^{240}Pu	3.9 %	-	2.6 %	-	3.3 %
	^{241}Am	27.6 %	37.3 %	19 %	36.1 %	29.5 %
	^{242}Cm	-	-	2.2 %	-	2.0 %
	^{244}Cm	26.7 %	35.7 %	32.8 %	34.4 %	21.6 %
T=0.5 y	Tot (W/g)	0.076	0.16	0.034	0.11	0.0005
Fabrication plants	^{238}Pu	31.9 %	18.8 %	26.7 %	27.0 %	29.8 %
	^{239}Pu	-	-	-	-	3.6 %
2150	^{240}Pu	-	-	2.2 %	-	5.2 %
	^{241}Am	-	4.0 %	4.7 %	3.9 %	7.3 %
	^{244}Cm	56.5 %	74.6 %	63.4 %	63.1 %	51.8 %
τ=0 y	Tot (W/g)	0.056	0.044	0.042	0.045	0.02
	^{238}Pu	40.3 %	46.5 %	40.8 %	47.2 %	41.1 %
	^{239}Pu	-	-	-	-	5.7 %
Repository	^{240}Pu	4.0%	3.3 %	5.1 %	3.3 %	8.8 %
	^{241}Am	21.6 %	29.8 %	23.8 %	28.6 %	28.2 %
	^{244}Cm	30.3 %	16.7 %	25.8 %	17.1 %	13.3 %
τ =100 y	Tot (W/g)	0.026	0.025	0.021	0.0.25	0.012
	^{238}Pu	40.6 %	38.8 %	37.9 %	40.0 %	31.5 %
	^{239}Pu	2.3 %	3.2 %	3.6 %	3.2 %	-
Repository	^{240}Pu	8.6 %	6.0 %	10.2 %	6.1 %	14.2 %
	^{241}Am	43.6 %	49.1 %	44.1 %	47.8 %	41.7 %
	^{242}Cm	2.0 %	-	-	-	
τ =1 000 y	Tot (W/g)	0.0056	0.0052	0.0052	0.0052	0.004
	^{239}Pu	10.5 %	14.6 %	14.3 %	14.9 %	27.6 %
Repository	^{240}Pu	35.9 %	25.8 %	37.9 %	26.4 %	39.3 %
	^{241}Am	47.9 %	54.9 %	43.3 %	54.1 %	30.2 %
	^{243}Am	3.3 %	3.2 %	2.7 %	3.2 %	-

Note: These results should be compared with those of Table 13. A few more isotope contributions are shown.

2.3.3.2 Results and discussion

Tables 10-13 show the decay heat, ingestion radiotoxicity, spontaneous fission rate and gamma dose rate resulting from the decay calculations of nuclide mixtures due to scenario calculations (here not reported). Decay heat and radiotoxicity are evaluated in the fabrication plant (decay time 0.5 y) and in the repository (after 0, 100, 1 000, 10 000, and 100 000 years). Spontaneous fission rate and gamma dose rate are evaluated only in the fabrication plant, where they can potentially pose safety and operational issues for workers. In all tables, percent values are shown which refer to 1 gram of reactor fuel discharged in ~2050 and ~2150, respectively.

Table 10: Decay heat nuclides share in reactor discharged fuel, fabrication plant and repository after 100 and 1 000 years

Case:		CR=0.5		CR=0.8		CR=1
Decay heat (α+β+γ)		MA/Pu=0.1	MA/Pu=1	MA/Pu=0.1	MA/Pu=1	MA/Pu=0.1
T=30 d	Tot (W/g)	0.35	0.41	0.17	0.29	0.087
FR discharge	^{242}Cm	83.8 %	76.4 %	84.3 %	76.8 %	88.0 %
2050	^{244}Cm	-	12.5 %	-	11.8 %	-
T=30 d	Tot (W/g)	0.37	0.35	0.12	0.23	0.17
FR discharge	^{238}Pu	-	16.9 %	-	17.7 %	-
2150	^{242}Cm	79.4 %	68.7 %	68.5 %	67.7 %	75.3 %
	^{244}Cm	11.4 %	12.3 %	22.0 %	12.4 %	14.3 %
T=0.5 y	Tot (W/g)	0.023	0.047	0.021	0.033	0.009
Fabrication plants	^{238}Pu	36.9 %	24 %	40.6 %	26.1 %	40.3%
2050	^{241}Am	27.6 %	37.3 %	19 %	36.1 %	29.5 %
	^{244}Cm	26.7 %	35.7 %	32.8 %	34.4 %	21.6 %
T=0.5 y	Tot (W/g)	0.076	0.16	0.034	0.11	0.005
Fabrication plants	^{238}Pu	31.9 %	18.8 %	26.7 %	27.0 %	29.8 %
2150	^{244}Cm	56.5 %	74.6 %	63.4 %	63.1 %	51.8 %
	Tot (W/g)	0.056	0.044	0.042	0.045	0.02
τ=0 y	^{238}Pu	40.3 %	46.5 %	40.8 %	47.2 %	41.1 %
Repository	^{241}Am	21.6 %	29.8 %	23.8 %	28.6 %	28.2 %
	^{244}Cm	30.3 %	16.7 %	25.8 %	17.1 %	13.3 %
	Tot (W/g)	0.026	0.025	0.021	0.025	0.012
τ =100 y	^{238}Pu	40.6 %	38.8 %	37.9 %	40.0 %	31.5 %
Repository	^{240}Pu	-	-	10.2 %	-	14.2 %
	^{241}Am	43.6 %	49.1 %	44.1 %	47.8 %	41.7 %
	Tot (W/g)	0.0056	0.0052	0.0052	0.0052	0.004
τ =1 000 y	^{239}Pu	10.5 %	14.6 %	14.3 %	14.9 %	27.6 %
Repository	^{240}Pu	35.9 %	25.8 %	37.9 %	26.4 %	39.3 %
	^{241}Am	47.9 %	54.9 %	43.3 %	54.1 %	30.2 %

Note: Decay heat nuclides share (absolute total value and %) in reactor discharged fuel (at the end of the scenario, after 30 days), in fabrication plant and repository after 100 and 1 000 years. The largest value is indicated in **bold** (Note: differences between the value of fabrication and repository at time 0 – i.e. end of the scenario – are due to different ^{238}Pu share, in particular for CR=1 they are equal to 0.29% in fabrication plant in 2150 and 1.43% in repository at time 0. This effect is due to the fact that in repository all reprocessing losses over the full duration of the scenario are accumulated, with consequent ^{242}Cm decay).

Table 11: Ingestion radiotoxicity nuclides share in reactor discharged fuel and repository after 1 000, 10 000, and 100 000 years

Case: Radiotoxicity		CR=0.5		CR=0.8		CR=1
		MA/Pu=0.1	MA/Pu=1	MA/Pu=0.1	MA/Pu=1	MA/Pu=0.1
τ=1 000 y	Tot (Sv/g)	1.44E+03	1.32E+03	1.35E+03	1.31E+03	1.09E+03
Repository	^{239}Pu	12.1 %	17.2 %	16.3 %	17.5 %	30.2 %
	^{240}Pu	41.6 %	30.4 %	43.1 %	31.0 %	42.9 %
	^{241}Am	41.4 %	48.4 %	36.8 %	47.6 %	24.7 %
τ=10 000 y	Tot (Sv/g)	4.11E+02	3.63E+02	4.28E+02	3.66E+02	4.49E+02
	^{239}Pu	34.8 %	50.3 %	41.2 %	50.3 %	57.1 %
Repository	^{240}Pu	56.5 %	42.7 %	52.7 %	42.8 %	40.2 %
τ=100 000 y	Tot (Sv/g)	3.53E+01	3.67E+01	3.27E+01	3.72E+01	3.0E+01
	^{210}Pb	15.3 %	14.1 %	13.1 %	14.3 %	-
Repository	^{210}Po	26.7 %	24.5 %	22.8 %	24.8 %	13.7 %
	^{239}Pu	32.7 %	39.4 %	42.4 %	39.1 %	65.3 %

Note: Ingestion radiotoxicity nuclides share (absolute total value in %) in reactor discharged fuel (at the end of the scenario, after 30 days) and repository after 1 000, 10 000, and 100 000 years. The largest value is indicated in **bold**.

Table 12: Spontaneous fission rate nuclides share in reactor discharged fuel and fabrication plant

Case: Spontaneous fission rate		CR=0.5		CR=0.8		CR=1
		MA/Pu=0.1	MA/Pu=1	MA/Pu=0.1	MA/Pu=1	MA/Pu=0.1
T=30 d	Tot ($s^{-1}\cdot g^{-1}$)	5.62E+04	9.33E+04	2.43E+04	6.31E+04	1.05E+04
FR discharge	^{242}Cm	31.6 %	20.3 %	37.2 %	21.4 %	44.7 %
2050	^{244}Cm	67.3 %	79.1 %	62.0 %	78.0 %	54.7 %
T=30 d	Tot ($s^{-1}\cdot g^{-1}$)	8.42E+04	7.90E+04	4.70E+04	5.32E+04	4.61E+03
FR discharge	^{242}Cm	21.6 %	18.7 %	10.6 %	18.3 %	17.1 %
2150	^{244}Cm	73.6 %	79.6 %	81.3 %	80.1 %	77.3 %
T=0.5 y	Tot ($s^{-1}\cdot g^{-1}$)	9.38E+03	2.48E+04	9.97E+03	1.70E+04	2.96E+03
Fabrication plant 2050	^{244}Cm	97.4 %	99.0 %	98.0 %	98.9 %	97.9 %
T=0.5 y	Tot ($s^{-1}\cdot g^{-1}$)	6.95E+04	1.88E+05	3.37E+04	1.34E+05	4.50E+03
Fabrication plant 2150	^{244}Cm	90.3 %	93.5 %	90.9 %	92.9 %	92.4 %

Note: Spontaneous fission rate nuclides share (absolute total value in %) in reactor discharged fuel (at the end of the scenario) and fabrication plant. The largest value is indicated in **bold**.

Table 13: Gamma dose rate nuclides share % in reactor discharged fuel and fabrication plant

Case: Gamma dose rate		CR=0.5		CR=0.8		CR=1
		MA/Pu=0.1	MA/Pu=1	MA/Pu=0.1	MA/Pu=1	MA/Pu=0.1
T=30 d	Tot (μSv/(h·g))	8.13E+01	1.08E+02	4.70E+01	7.95E+01	2.08E+01
	^{239}Np	-	-	10.3 %	-	17.8 %
FR discharge	^{241}Am	20.3 %	33.6 %	24.1 %	33.2 %	29.2 %
2050	^{242}Am	20.3 %	20.6 %	20.8 %	20.7 %	17.8 %
	^{242m}Am	26.2 %	26.6 %	26.8 %	26.7 %	23.0 %
	^{243}Cm	21.7 %	-	14.9 %	-	-
T=30 d	Tot (μSv/(h·g))	8.67E+01	8.65E+01	3.02E+01	5.91E+01	6.92E+00
	^{239}Np	-	-	19.3 %	-	52.9 %
FR discharge	^{241}Am	19.9 %	32.7 %	19.6 %	31.7 %	14.9 %
	^{242}Am	20.5 %	20.2 %	18.3 %	20.1 %	11.2 %
2150	^{242m}Am	26.4 %	26.0 %	23.6 %	25.9 %	14.4 %
	^{243}Cm	21.4 %	-	15.1 %	-	-
T=0.5 y	Tot (μSv/(h·g))	4.45E+01	1.06E+02	3.39E+01	7.37E+01	1.95E+01
Fabrication	^{241}Am	69.9 %	79.8 %	55.0 %	79.0 %	66.7 %
plants	^{242m}Am	-	-	17.0 %	-	13.1 %
2050	^{242}Am	-	-	13.2 %	-	10.2 %
T=0.5 y	Tot (μSv/(h·g))	7.21E+01	1.29E+02	2.56E+01	8.95E+01	5.16E+00
	^{241}Am	35.3 %	24.1 %	29.7 %	23.7 %	37.6 %
Fabrication	^{242m}Am	22.3 %	30.9 %	25.8 %	31.2 %	25.0 %
plants	^{242}Am	17.3 %	23.9 %	20.0 %	24.2 %	19.4 %
2150	^{243}Cm	17.3 %	10.4 %	15.2 %	10.8 %	-

Note: Gamma dose rate nuclides share (absolute total value in %) in reactor discharged fuel (at the end of the scenario, after 30 days) and fabrication plant. The largest value is indicated in **bold**.

Direct comparison of obtained results with the reference values published in [21] is not easy due to different assumptions (e.g. SF cooling times etc.). However, values for respectively decay heat and neutron source as reported in [21] were compared with the results of the present investigation and the orders of magnitude seem to agree reasonably. Also, general trends seem to be in agreement (in particular the increase in heat load and neutron emission originated from a core with CR=1 to CR=0.5 with a large MA content).

Table 14 shows the mass flow rates for the CR and MA/Pu cases quoted above in fabrication plants in ~2050 and ~2150 and in repository at the end of the scenario. These mass flows are consistent with the burner reactor deployment given in Figures 13, 16 and 19. According to the different scenario objectives, as explained above, different inventories are shown not only for different MA/TRU ratios in the fuel, but also for different conversion ratio values. Finally, the low flow in repository is due to the fact that only losses (0.1%) are stored in the geological repository.

Table 14: Scenario masses flow rate in fabrication plant and repository: mass, decay heat, spontaneous fission and gamma dose total values (the larger value in the row in bold)

Scenario flow masses (annual inputs - t/y)	CR=0.5		CR=0.8		CR=1
	MA/Pu=0.1	MA/Pu=1	MA/Pu=0.1	MA/Pu=1	MA/Pu=0.1
Fabrication plant (2050)	59	9	137	14	185
Fabrication plant (2150)	152	23	378	36	508
Repository (2200)	0.09	0.03	0.1	0.03	0.1

Table 15 summarises the results inferred from Tables 10-13: ^{241}Am and ^{244}Cm are relevant for decay heat and ingestion radiotoxicity aspects, ^{244}Cm for spontaneous fission, and ^{241}Am, ^{242m}Am, ^{242}Am (coming from ^{242m}Am decay), and ^{243}Cm for gamma dose rate in fabrication plant; it should be concluded that nuclides relevant in close fast critical scenarios are:

- ^{241}Am;
- ^{242m}Am;
- ^{243}Cm;
- ^{244}Cm.

Table 15: Summary of the most relevant nuclide effects on the fuel cycle

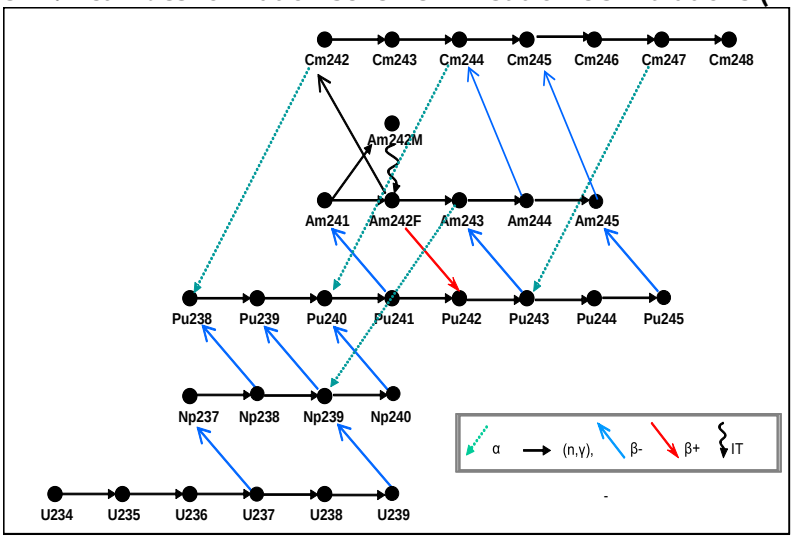
Relevant nuclides	Heat load	Ingestion radiotoxicity	Spontaneous fission rate	Gamma dose rate
^{241}Am	X	X		X
^{242}Am				X
^{242m}Am				X
^{243}Cm				X
^{244}Cm	X		X	

Moreover, from an analysis of the same tables it can be concluded that the core design which presents the higher specific values is the **CR=0.5 – MA/Pu=1** configuration, in agreement with the conclusions drawn in [21].

As expected, the same nuclides are relevant in the case of an ADS with a similar fuel loading (e.g. in terms of MA/TRU ratio).

To provide a preliminary indication of the important reactions which cause the build-up of nuclides quoted above, a build-up scheme is shown in Figure 22. In particular, ^{241}Am (produced by ^{241}Pu β-decay, half-life = 14.35 y) by neutron capture produces ^{242m}Am (~10%) – which builds up - and ^{242}Am (~90%) (half-life, respectively, 141 y and 16.04 h), which decays with a branching ratio of 82.7% into ^{242}Cm (17.3% decays into ^{242}Pu, which by neutron capture decays into ^{243}Pu, which in its turn decays with a half-life of 4.956 h into ^{243}Am), which by successive neutron captures allows the formation of ^{244}Cm. Another route is the neutron capture of ^{242}Am which produces ^{243}Am (half-life=7370 y), which by capture becomes ^{244}Am (half-life=10.1 h) decaying by β decay into ^{244}Cm. As fission reaction could play a role in the nuclides removal too, it can be concluded that relevant reactions which influence the formation of critical nuclides in fuel cycles are capture and fission of ^{241}Am, ^{242}Am, ^{243}Am, ^{244}Am, ^{242}Cm, ^{243}Cm and ^{244}Cm. A further sensitivity and uncertainty analysis would be required in order to provide quantified uncertainty contributions by isotope and reaction type.

Figure 22: Actinides formation scheme in neutronic simulations (KIT)



2.3.3.3 Conclusions

A set of scenario simulations of different fuel cycles options based on critical fast reactors with various conversion ratios and fuelled with different minor actinides to plutonium ratios was assessed. The most important contributors to crucial fuel cycle parameters, in particular heat load, ingestion radiotoxicity, neutron and gamma emission rate in fuel fabrication plant and repository, were investigated. It was shown that, despite the significant differences of the scenarios, only a restricted number of MA and Pu isotopes give a significant contribution to the parameters mentioned above.

The most important are ^{241}Am (both for heat load and radiotoxicity) and ^{244}Cm (in particular for the neutron emission rate), although ^{242m}Am and ^{243}Cm provide a sensible impact too.

It should be noted that all reactions taking to build-up of these nuclides and to their removal (such as capture or fission) are of interest, therefore research should focus also on fathers of the most problematic radionuclides: this means that capture and fission reactions of ^{241}Am, ^{242}Am, ^{243}Am, ^{244}Am, ^{242}Cm, ^{243}Cm and ^{244}Cm should be investigated in principle, although a careful assessment of these reactions should be preliminary performed. The behaviour of the fast decaying nuclides such as ^{242}Cm should also be investigated, as they represent the major source of ^{238}Pu in the fuel cycle, which, in its turn, represents one of the most relevant isotopes from the heat load point of view. Nevertheless, the assessment of fast decaying nuclides cross-sections will be probably difficult and not practical due to handling/transport difficulties and high cost.

Moreover, it was possible to show that the CR=0.5 – MA/Pu=1 core configuration (representative of MA burner FRs) is the most critical one, according to results already available in literature.

It is worth noting here that in the scenario studies the cross-section uncertainties represent only one factor which affects global uncertainties among many others (e.g. cooling times, start-up dates, fleet introduction pace, etc.). Future work will focus on the impact of cross-sections uncertainties on the fuel cycle parameters adopted in the present chapter.

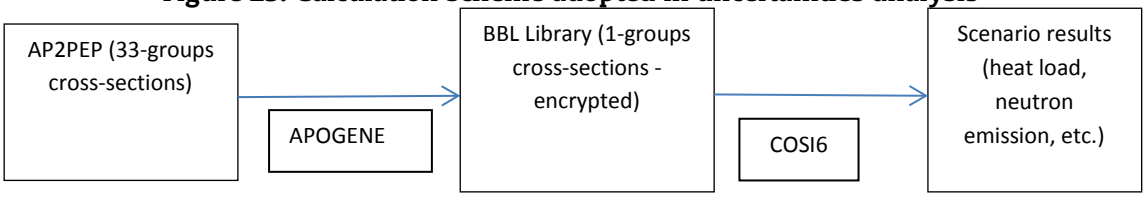
2.3.4 Nuclear fuel cycle scenario uncertainties analysis

2.3.4.1 Assumptions and calculation procedure

According to the results presented in Section 2.3.3, the same scenario scheme presented in Figure 11 and discussed in Section 2.3.3.1 was adopted: The most representative configuration (i.e. a burner critical fast reactor with a high-MA content in the fuel), i.e. the CR=0.5 and MA/Pu=1 core configuration was considered for uncertainties analysis evaluation.

The scenario calculations for this and earlier studies [22, 23] were performed with cross-section libraries generated at KIT earlier [24] by employing the ERANOS code [13]. The ERANOS output, AP2PEP file was processed with APOGENE [25] software in order to collapse the 33-group to 1-group cross-sections, to interpolate and to format them as required by the COSI6 (ver.6.0.1) [26] scenario code in order to evaluate scenario results according to the procedure discussed in the previous paragraph. Figure 23 shows the flow scheme of the entire process.

Figure 23: Calculation scheme adopted in uncertainties analysis



Concerning the FR system, a low CR core was adopted, with a MA/Pu ratio ~1 (basic nuclear data based on JEF2.2 cross-section library). Further details about the core characteristics can be found in [22, 23].

The 1-group cross-sections are used in COSI6. The output of the scenario code are, as previously indicated, the compositions and mass flows in reactors, fabrication and reprocessing plants, etc. vs. time. In order to evaluate the relevant quantities as heat load or neutron emission, the compositions are extracted from the COSI6 output and processed with Nucleonica Decay Engine [27].

For the present analysis, the cross-section sensitivity coefficients needed for the uncertainty analysis, were obtained with a direct method by increasing and decreasing each isotope and reaction cross-section values by 5%. For this purpose, a code (KAPAN, [28]) was developed which reads the AP2PEP and increases/decreases the cross-sections for a user-defined nuclide and reaction by a user-defined amount at every burn-up step.

In particular, both capture and fission cross-sections were increased/decreased, according to the following scheme:

- Capture cross-sections:

$$\sigma'_{ABS} = \sigma_{ABS} \pm p \cdot (\sigma_{ABS} - \sigma_{FIS}) \qquad (1)$$

where σ designates the original microscopic cross-section value (ABS – absorption, FIS – fission), p is the introduced change and σ' is the new value.

- Fission cross-sections:

$$\sigma'_{FIS} = (1 \pm p) \cdot \sigma_{FIS} \qquad (2)$$

$$\sigma'_{ABS} = \sigma_{ABS} \pm p \cdot \sigma_{FIS} \qquad (3)$$

where σ designates the original microscopic cross-section value (ABS – absorption, FIS – fission), p is the introduced change and σ' is the new value.

KAPAN code provides an output, among others, also the 1-group collapsed cross-sections (useful for checking purposes) and reaction rates - averaged over all the burn-up steps - per every energy group according to the following formulas:

$$f_i = \frac{\sigma_i \cdot \phi_i}{\sum_i \sigma_i \cdot \phi_i} \quad (4)$$

$$f_{i,mean} = \frac{\sum_B f_{i,B}}{N_B} \quad (5)$$

where f_i is the reaction rate percentage factor, σ_i stands for microscopic cross-section and ϕ_i for neutron flux (all three for neutrons with energy corresponding to group i), $f_{i,mean}$ is the reaction rate percentage factor for neutrons with energy corresponding to group i averaged over all burn-up points B, N_B is the number of burn-up points.

2.3.4.2 Analysis results

- 2.3.4.2.1 Sensitivity coefficients calculation

For the present sensitivity analysis, four parameters have been considered:

- heat load at:
  - FR fabrication plant in 2050;
  - FR reactor discharge in 2149 (after 30 days cooling);
  - repository in 2199.
- Neutron emission:
  - FR fabrication plant in 2050.

FR systems are introduced in the scenario at a constant pace starting from 2035 until 2080, and then a constant energy demand is assumed (Figures 12 and 13).

In order to evaluate the sensitivity coefficients, a reference value was considered for unperturbed case (R_0), and then perturbed values for a fixed cross-section variation c (5% in the present case) were evaluated according to the following equations:

$$S^+ = \frac{R_0^+ - R_0}{c \cdot R_0} \quad (6)$$

$$S^- = \frac{R_0^- - R_0}{-c \cdot R_0} \quad (7)$$

Table 16 shows the summary of the results obtained applying a perturbation value of 5% to the absorption and fission cross-sections of the following nuclides for the reactor system considered: ^{235}U, ^{238}U, ^{237}Np, ^{238}Pu to ^{242}Pu, ^{241}Am, ^{242m}Am, ^{243}Am, ^{242}Cm to ^{245}Cm (15 actinides).

Table 16: Sensitivity coefficients

Nuclide	Sensit. Coeff.	Capture				Fission			
		Heat load			Neutron emission	Heat load			Neutron emission
		FR fabrication (2050)	FR reactor discharge (2149)	Repository (2199)	FR fabrication (2050)	FR fabrication (2050)	FR reactor discharge (2149)	Repository (2199)	FR fabrication (2050)
^{235}U	S^-	-3.625E-08	1.319E-04	-3.910E-05	9.422E-08	-2.540E-06	-2.790E-03	-5.466E-04	6.601E-06
	S^+	1.782E-07	4.181E-04	8.398E-05	-4.632E-07	-2.258E-06	-2.368E-03	-3.997E-04	5.869E-06
^{238}U	S^-	-5.689E-05	-5.594E-03	-1.073E-01	1.478E-04	-4.997E-05	-5.162E-02	-6.108E-03	1.299E-04
	S^+	-5.588E-05	-2.733E-03	-1.348E-01	1.452E-04	-4.946E-05	-5.099E-02	-6.031E-03	1.285E-04
^{237}Np	S^-	2.997E-04	1.201E-01	1.472E-01	-7.788E-04	3.943E-05	-7.154E-03	8.341E-04	-1.025E-04
	S^+	2.929E-04	1.143E-01	1.146E-01	-7.611E-04	3.907E-05	-6.956E-03	8.543E-04	-1.015E-04
^{238}Pu	S^-	-2.276E-06	-3.173E-02	-6.471E-02	5.916E-06	-6.759E-05	-2.200E-01	-1.048E-01	1.756E-04
	S^+	-2.051E-06	-3.082E-02	-6.318E-02	5.330E-06	-6.564E-05	-2.096E-01	-1.288E-01	1.706E-04
^{239}Pu	S^-	7.715E-06	6.889E-03	5.924E-04	-2.005E-05	-2.345E-04	-1.366E-01	1.148E-01	6.094E-04
	S^+	7.918E-06	7.077E-03	1.050E-03	-2.058E-05	-2.248E-04	-1.336E-01	7.346E-02	5.843E-04
^{240}Pu	S^-	-2.347E-05	4.453E-03	1.437E-02	6.100E-05	-5.760E-05	-4.621E-02	2.231E-02	1.497E-04
	S^+	-2.310E-05	4.565E-03	1.405E-02	6.004E-05	-5.662E-05	-4.413E-02	2.232E-02	1.472E-04
^{241}Pu	S^-	2.228E-06	-1.102E-03	4.590E-04	-5.790E-06	-1.534E-05	-2.534E-02	1.168E-03	3.987E-05
	S^+	2.428E-06	-8.027E-04	5.841E-04	-6.311E-06	-1.459E-05	-2.384E-02	1.203E-03	3.792E-05
^{242}Pu	S^-	-1.577E-05	-7.986E-03	1.516E-02	4.099E-05	-1.060E-05	-1.028E-02	-1.455E-03	2.754E-05
	S^+	-1.547E-05	-7.546E-03	1.468E-02	4.021E-05	-1.031E-05	-9.609E-03	-1.275E-03	2.679E-05
^{241}Am	S^-	1.239E-04	4.337E-01	5.211E-02	-3.220E-04	-1.036E-05	7.868E-02	9.086E-03	2.693E-05
	S^+	1.205E-04	4.091E-01	3.124E-02	-3.132E-04	-1.024E-05	7.975E-02	9.307E-03	2.661E-05
^{242m}Am	S^-	-4.080E-07	-2.255E-03	4.180E-03	1.060E-06	4.305E-06	-1.010E-02	1.658E-03	-1.119E-05
	S^+	-1.895E-07	-1.932E-03	4.273E-03	4.924E-07	4.364E-06	-9.113E-03	2.090E-03	-1.134E-05
^{243}Am	S^-	2.851E-05	1.073E-01	1.600E-01	-7.409E-05	1.606E-05	-1.077E-03	-3.306E-02	-4.173E-05
	S^+	2.818E-05	1.038E-01	1.539E-01	-7.323E-05	1.615E-05	-7.904E-04	-3.261E-02	-4.197E-05
^{242}Cm	S^-	-1.298E-06	-8.926E-03	3.176E-04	3.372E-06	-1.925E-06	-2.146E-02	-1.944E-03	5.004E-06
	S^+	-1.084E-06	-8.627E-03	4.452E-04	2.816E-06	-1.708E-06	-2.110E-02	-1.813E-03	4.439E-06
^{243}Cm	S^-	-1.123E-07	-1.372E-04	7.073E-05	2.917E-07	1.915E-07	-1.422E-03	-1.309E-03	-4.977E-07
	S^+	1.046E-07	1.495E-04	1.983E-04	-2.719E-07	3.944E-07	-1.058E-03	-1.107E-03	-1.025E-06
^{244}Cm	S^-	-2.373E-06	-7.994E-03	-6.463E-02	6.166E-06	6.555E-06	-1.793E-02	-6.571E-02	-1.704E-05
	S^+	-2.144E-06	-7.676E-03	-6.328E-02	5.571E-06	6.686E-06	-1.745E-02	-6.460E-02	-1.738E-05
^{245}Cm	S^-	-2.704E-07	-8.157E-04	3.486E-04	7.027E-07	4.856E-06	-4.339E-03	1.013E-02	-1.262E-05
	S^+	-5.910E-08	-5.168E-04	4.762E-04	1.536E-07	4.895E-06	-3.705E-03	1.059E-02	-1.272E-05

An example (referred to ^{241}Am capture cross-section) of the sensitivity coefficients splitting in energy groups according to equations 4) and 5) is presented in Table 17. Sum values (in **bold**) are those reported in Table 16.

Table 17: Sensitivity coefficients split in energy groups according to evaluated reaction rates

^{241}Am (Capture)		Heat load						Neutron emission	
Energy group	Reaction rates	FR Fabr. 2050	FR Fabr. 2050	FR reactor discharge (2149)	FR reactor discharge (2149)	Repository (2199)	Repository (2199)	FR fabrication (2050)	FR fabrication (2050)
		S+	S-	S+	S-	S+	S-	S+	S-
1	1.322E-06	1.64E-10	1.59E-10	-4.3E-10	-4.1E-10	6.889E-08	4.130E-08	5.733E-07	5.408E-07
2	2.267E-05	2.81E-09	2.73E-09	-7.30E-09	-7.1E-09	1.181E-06	7.082E-07	9.832E-06	9.274E-06
3	1.703E-04	2.11E-08	2.05E-08	-5.5E-08	-5.3E-08	8.872E-06	5.318E-06	7.383E-05	6.964E-05
4	6.518E-04	8.08E-08	7.85E-08	-2.1E-07	-2E-07	3.397E-05	2.036E-05	2.827E-04	2.666E-04
5	2.572E-03	3.19E-07	3.1E-07	-8.3E-07	-8.1E-07	1.340E-04	8.035E-05	1.115E-03	1.052E-03
6	1.717E-02	2.13E-06	2.07E-06	-5.5E-06	-5.4E-06	8.947E-04	5.363E-04	7.445E-03	7.023E-03
7	6.347E-02	7.86E-06	7.65E-06	-2E-05	-2E-05	3.307E-03	1.983E-03	2.752E-02	2.596E-02
8	8.739E-02	1.08E-05	1.05E-05	-2.8E-05	-2.7E-05	4.554E-03	2.730E-03	3.790E-02	3.575E-02
9	1.043E-01	1.29E-05	1.26E-05	-3.4E-05	-3.3E-05	5.437E-03	3.259E-03	4.524E-02	4.268E-02
10	1.416E-01	1.75E-05	1.71E-05	-4.6E-05	-4.4E-05	7.377E-03	4.422E-03	6.139E-02	5.791E-02
11	1.370E-01	1.7E-05	1.65E-05	-4.4E-05	-4.3E-05	7.141E-03	4.281E-03	5.942E-02	5.605E-02
12	9.053E-02	1.12E-05	1.09E-05	-2.9E-05	-2.8E-05	4.718E-03	2.828E-03	3.926E-02	3.703E-02
13	9.613E-02	1.19E-05	1.16E-05	-3.1E-05	-3E-05	5.010E-03	3.003E-03	4.169E-02	3.932E-02
14	8.758E-02	1.09E-05	1.06E-05	-2.8E-05	-2.7E-05	4.564E-03	2.736E-03	3.798E-02	3.583E-02
15	4.012E-02	4.97E-06	4.83E-06	-1.3E-05	-1.3E-05	2.091E-03	1.253E-03	1.740E-02	1.641E-02
16	2.449E-02	3.03E-06	2.95E-06	-7.9E-06	-7.7E-06	1.276E-03	7.651E-04	1.062E-02	1.002E-02
17	1.565E-02	1.94E-06	1.89E-06	-5E-06	-4.9E-06	8.155E-04	4.889E-04	6.787E-03	6.402E-03
18	7.975E-03	9.88E-07	9.61E-07	-2.6E-06	-2.5E-06	4.156E-04	2.491E-04	3.458E-03	3.262E-03
19	2.572E-02	3.19E-06	3.1E-06	-8.3E-06	-8.1E-06	1.340E-03	8.033E-04	1.115E-02	1.052E-02
20	1.833E-02	2.27E-06	2.21E-06	-5.9E-06	-5.7E-06	9.553E-04	5.726E-04	7.949E-03	7.499E-03
21	1.168E-02	1.45E-06	1.41E-06	-3.8E-06	-3.7E-06	6.085E-04	3.648E-04	5.064E-03	4.777E-03
22	5.245E-03	6.5E-07	6.32E-07	-1.7E-06	-1.6E-06	2.733E-04	1.639E-04	2.275E-03	2.146E-03
23	8.641E-03	1.07E-06	1.04E-06	-2.8E-06	-2.7E-06	4.503E-04	2.699E-04	3.747E-03	3.535E-03
24	4.755E-03	5.89E-07	5.73E-07	-1.5E-06	-1.5E-06	2.478E-04	1.485E-04	2.062E-03	1.945E-03
25	1.124E-03	1.39E-07	1.35E-07	-3.6E-07	-3.5E-07	5.856E-05	3.511E-05	4.873E-04	4.597E-04
26	2.545E-03	3.15E-07	3.07E-07	-8.2E-07	-8E-07	1.326E-04	7.949E-05	1.104E-03	1.041E-03
27	2.173E-03	2.69E-07	2.62E-07	-7E-07	-6.8E-07	1.132E-04	6.789E-05	9.424E-04	8.890E-04
28	8.530E-04	1.06E-07	1.03E-07	-2.7E-07	-2.7E-07	4.445E-05	2.665E-05	3.699E-04	3.489E-04
29	6.656E-04	8.25E-08	8.02E-08	-2.1E-07	-2.1E-07	3.468E-05	2.079E-05	2.886E-04	2.723E-04
30	3.855E-04	4.78E-08	4.64E-08	-1.2E-07	-1.2E-07	2.009E-05	1.204E-05	1.672E-04	1.577E-04
31	6.967E-04	8.63E-08	8.4E-08	-2.2E-07	-2.2E-07	3.631E-05	2.176E-05	3.021E-04	2.850E-04
32	3.317E-04	4.11E-08	4E-08	-1.1E-07	-1E-07	1.729E-05	1.036E-05	1.439E-04	1.357E-04
33	1.201E-05	1.49E-09	1.45E-09	-3.9E-09	-3.8E-09	6.258E-07	3.751E-07	5.207E-06	4.912E-06
SUM	1.00E+00	1.239E-04	1.205E-04	4.337E-01	4.091E-01	5.211E-02	3.124E-02	-3.220E-04	-3.132E-04

- 2.3.4.2.2 Uncertainty analysis

In order to perform the uncertainty analysis, the S^+ and S^- sensitivity coefficients (by isotope, reaction, and energy group) were averaged. This averaging corresponds to adopt the central differentiation formula:

$$\boldsymbol{S} = (\boldsymbol{V}^+ - \boldsymbol{V}^-)/2\Delta \qquad (8)$$

where S is the sensitivity coefficient, and V^+ and V^- are the values corresponding to a $\pm\Delta$ variation (in this case Δ =5%) of the parameter of which one can calculate the sensitivity coefficients. The central differentiation formula helps to smooth out possible non-linearity. Then, the uncertainty sandwich formula was applied where sensitivity coefficients are folded with covariance data:

$$\boldsymbol{U^2} = \boldsymbol{S^T C\, S} \qquad (9)$$

U is the total uncertainty, S^T is the transposed of the sensitivity array, and *C* is the covariance matrix of the cross-section. The covariance matrix contains standard deviation on the diagonal and correlations among cross-sections in the off-diagonal terms. The COMMARA 2.0 covariance matrix [6] used in this study is not consistent with the JEF2.2 nuclear data used, however, this seems acceptable for this type of scoping evaluations [29].

Tables 18 and 19 show the uncertainties obtained for the heat load parameters at reactor discharge in year 2149 and at repository in year 2199. No significant uncertainties were obtained for the other two parameters (heat load and neutron emission at fabrication in year 2050) given the very small associated sensitivity coefficients. In the tables, uncertainties are broken down by isotopes and reactions. Sums are performed in a statistical way (square root of sum of square values with correlations taken into account), and only contributions greater than 0.1% uncertainty are reported.

For the heat load at reactor discharge in year 2149, the major contribution is given by the fission reaction of ^{242}Cm followed by the capture of ^{241}Am. For the heat load at repository in year 2199, the major contribution comes from the capture of ^{244}Cm and then by the capture of ^{238}Pu and ^{243}Am, while for the fission reaction only that of ^{244}Cm has a significant value. However, the total uncertainty in both cases is very small and would not represent a concern in fuel cycle assessments. Nevertheless, one has to be very cautious in drawing conclusions. These are quite complex calculations and there is always the possibility that something significant could be neglected. For instance, it is not clear if a full explicit calculation of equilibrium densities is performed at the level of the COSI code. One evidence for this statement is the relatively low computed sensitivity to the ^{238}U capture that normally tends to dominate in this kind of calculation as it is at the cross road of the burn-up chain. Moreover, fission products have not been taken into consideration in this uncertainty analysis for the heat load. If they were considered, besides the obvious contribution from the fission products, one would expect some large sensitivity to the minor actinide fission cross-sections that are at the origin of their production. Significant uncertainties are attached to minor actinide fission cross-sections.

Table 18: Uncertainty (expressed in %) for heat load at reactor discharge in year 2149

Isotope	σ_{cap}	σ_{fiss}	Sum
^{237}Np	0.5	0.0	0.5
^{238}Pu	0.5	0.5	0.7
^{239}Pu	0.0	0.1	0.1
^{240}Pu	0.0	0.1	0.1
^{242}Pu	0.2	0.0	0.2
^{241}Am	1.0	0.1	1.0
^{242m}Am	0.1	0.1	0.1
^{243}Am	0.6	0.0	0.6
^{242}Cm	0.6	1.8	1.9
^{244}Cm	0.3	0.2	0.4
Total	1.5	1.9	2.5

Table 19: Uncertainty (expressed in %) for heat load at repository in year 2199

Isotope	σ_{cap}	σ_{fiss}	Sum
^{238}U	0.1	0.0	0.1
^{237}Np	0.5	0.0	0.5
^{238}Pu	1.0	0.3	1.0
^{240}Pu	0.1	0.1	0.1
^{242}Pu	0.3	0.0	0.3
^{241}Am	0.1	0.0	0.1
^{243}Am	0.9	0.3	0.9
^{242}Cm	0.0	0.2	0.2
^{244}Cm	2.7	0.9	2.9
^{245}Cm	0.0	0.3	0.3
Total	3.1	1.1	3.3

- 2.3.4.2.3 Conclusions

The results obtained indicate that the present neutron cross-section uncertainties have a rather limited impact on the assessment of parameters of the fuel cycle such as the decay heat at different time horizons. This result is consistent with what was already pointed out in a previous study [1]. However, it should be noted that in the present study and in [1] only neutron cross-section uncertainties have been considered and no uncertainties on other nuclear parameters have been accounted for. For example, in the case of decay heat, only the decay heat associated with TRU, which is dominating at long cooling time, has been considered. No uncertainties on decay constants or fission yields have been considered that would have been associated to fission products (FP) decay heat. In the scenario considered in the present study, most of the TRU decay heat comes from isotopes like ^{242}Cm, ^{244}Cm, ^{241}Am, ^{238}Pu. The decay heat uncertainty is essentially related to the evaluation of the uncertainty of the inventory of these isotopes, which is due mostly to neutron cross-section uncertainty. Moreover, the production of FP is proportional to the fission cross-section of MA that carries significant uncertainty.

3. Existing integral data

3.1 Belgium

The programme of integral minor actinides nuclear data measurements in fast neutron spectrum core has been launched recently. In April-May 2011, some of the fission rates ratios of MA were measured in critical zero power fast core at VENUS-F Facility.

That time VENUS-F Facility was loaded by GUINEVERE core (FP6 EU Project), which simulated a fast critical core of an ADS, cooled by lead. The fuel of GUINEVERE core was uranium metal 30% enriched [30, 31, 32].

The fission rates ratios of ^{234}U, ^{237}Np, ^{240}Pu and ^{242}Pu to the ^{239}Pu were measured by small fission chambers in the centre and on the half of the radius of the GUINEVERE critical core. At the same places, the standard spectrum indexes as fission rates of ^{239}Pu and ^{238}U to the ^{235}U were measured. The analysis of the experiments by European partners is close to the finalisation.

In the next four years, the programme of MA integral data measurements will be foreseen in the frame of the on-going FP7 EU Project FREYA [33], which includes the investigations of the mock-ups of ADS MYRRHA (subcritical/critical) and LFR cores.

3.2 France

France has undertaken a comprehensive R&D effort to investigate the scientific and technical feasibility of various options for optimised plutonium and actinide management under the 1991 and 2006 *Sustainable Radioactive Waste Management Acts*. An essential consideration is the use of uranium and plutonium multi-recycling in fast neutron reactors, in order to limit the accumulation of sensitive materials, preserve natural resources, and simultaneously limit the amount of ultimate waste. In a long-term perspective, minor actinide separation and transmutation using these fast reactors could provide additional benefits. Recycling americium alone would reduce the footprint of the final waste repository by a factor of 10, while recycling americium and curium would reduce the long-term radiotoxicity of such waste by up to a factor of 100 [34].

CEA scenario studies applied to the French fleet of nuclear plants recommend a very progressive introduction of fast reactors, with a few units brought to operation around 2050 in a synergetic scheme of operation with thermal reactors, for optimal plutonium management. A more extensive deployment would come at a later stage, and could include minor actinide transmutation in a fast reactor fleet, with the construction of appropriate fuel cycle facilities. CEA studies have focused on americium multi-recycling to achieve a 99% elimination rate, using a uranium-oxide based transmutation fuel which can be reprocessed in the same facilities as driver fuels. Two options have been considered: “homogeneous recycling” and “heterogeneous recycling” of americium. In the homogeneous recycling, americium is diluted in the reactor fuel subassemblies, leading to an equilibrium concentration of about 1%, a low enough value to prevent the in-core fuel behaviour and performance from being significantly impacted. In the heterogeneous recycling, about 10% americium is added to UO_2 in blanket “target” subassemblies loaded at the periphery of the core.

CEA research has led to the successful development of laboratory-scale chemical processes for minor actinide separation (Am alone or Am+Cm together) from spent UOX fuels. At the same time, the capability of fast reactors to transmute americium efficiently has been established by experiments performed in the PHENIX reactor: the SUPERFACT experiment for the homogeneous option, and reduced-length pin irradiations for the heterogeneous option. However, much development work regarding transmutation fuel fabrication and processing, including larger-scale experimental demonstrations, is still needed before an industrial implementation can be considered.

The objective of the ASTRID sodium-cooled fast neutron demonstrator, currently in a preliminary design phase, is not only to prove the plutonium multiple recycling capability but also to perform demonstration experiments of minor actinide transmutation, at a scale ranging from a single fuel pin to a full-scale fuel subassembly.

From the reactor physics standpoint, one of the conclusions of the French studies is that minor actinide nuclear data have to be improved so that the corresponding uncertainties do not adversely impact plant safety, design or operation, including handling and transportation processes, in view of the fairly large inventories of these minor actinides that will be present in the fuel cycle. This is particularly important for ^{241}Am nuclear data, especially in the heterogeneous option, as this option implies much higher concentrations of this nuclide than in the homogeneous option. Improved ^{241}Am total, capture and fission cross-sections at intermediate and high neutron energies are required for accurate predictions of americium and daughter nuclide inventories, as well as other derived quantities. Of particular concern are ^{242}Cm and ^{244}Cm build-ups and the resulting increase in decay heat, which could seriously impair subassembly handling operations in the heterogeneous option. There is a similar concern over spontaneous neutron emission levels, before and after irradiation, which will be several orders of magnitude higher than for plutonium fuel. In addition, gas production also needs to be well predicted, as it impacts pin internal pressure.

In an attempt to improve americium nuclear data, CEA spearheaded a comprehensive experimental physics programme, aimed at producing a new evaluated ^{241}Am nuclear data file. Part of this programme consisted of a series of new challenging differential measurements, which were performed at the JRC/IRMM Geel facilities. These included in particular capture yield and transmission measurements, as well as high-resolution (n, 2n) reaction measurements. Another part of this programme consisted in irradiating small separated americium isotope samples (PROFIL-R and -M) in PHENIX; the corresponding post-irradiation examinations are still underway.

The new data from IRMM Geel were combined with earlier measurements, as well as with the limited number of relevant integral experiments, as inputs to nuclear model calculation codes. The tables in the last section provide a list of these integral experiments involving americium isotopes, performed in France, at the MASURCA fast critical facility (CIRANO, COSMO, MUSE-4) and in the PHENIX fast reactor (SUPERFACT, PROFIL and PROFIL-2). As a first outcome of this multi-year collaborative effort, a new, consistent evaluated ^{241}Am nuclear data file was assembled. This new evaluation, which includes covariance information, has recently been adopted for inclusion in the JEFF-3.2 library.

3.3 Italy

Experiment for the ^{242m}Am Production

The ^{242m}Am production through ^{241}Am(n; g) ^{242m}Am reaction was investigated [35]. The ratio between the number of ^{242m}Am and ^{241}Am atoms after an irradiation time t (initial condition: $N_{42m}(0) = 0$) is:

$$N_{42m}(t)/N_{41}(0)\exp(-R_{41}t) = b\, R_{41}\,[1-\exp(\lambda^{*}_{42m} - R_{41}t)]/(\lambda^{*}_{42m} - R_{41}) \qquad (10)$$

b = branching ratio = 0.175; R_{41} = ^{241}Am capture rate;

$\lambda^{*}_{42m}=\lambda_{42m}+ R_{42m,fis} + R_{42m,cpt}$ = ^{242m}Am "depletion" rate: λ_{42m} = 1.56E-10 sec^{-1}.

Four samples of ^{241}Am were prepared extracting Americium nitrate from (U,Pu) mixed oxides [36]. Drops of the solution were dried and deposited on nuclear grade Al substrate. The samples were grouped in two pairs; each of them includes a Cd-lined unit (1 mm thickness) in order to suppress the thermal neutron flux, avoiding the fissions of the ^{242m}Am just produced, and a bare one with the following ^{241}Am masses:

– bare samples: 0.299 ± 0.071 mg and 0.385 ± 0.006 mg;

– Cd-lined samples: 0.380 ± 0.009 mg and 0.460 ± 0.046 mg.

Before the neutron irradiation, the ^{242m}Am content was also measured by gamma-ray counting and found to be less than 1.5 ppm. Both pairs were placed in the central channel of the TRIGA RC-1 reactor [37], operating in an integral neutron flux of 2.68×10^{13} n/cm^{2}sec, with: $(\Phi_{th} + \Phi_{fast})/ \Phi_{fast} = 1.73$. The irradiation times were 30 hours and 119 hours for each pair, respectively.

After irradiation, the ^{241}Am and ^{242m}Am masses were determined measuring the sample activities by means of high resolution gamma-ray counting. A coaxial intrinsic Ge detector (r = 2.85 cm, h = 4.71 cm) was used with a nominal efficiency, at 1.33 MeV, of 25% with respect to a 3"x 3" NaI. For the determination of the produced ^{242m}Am mass, the intensities of the lines were measured at 984.5 keV, 1025.9 keV and 1028.5 keV emitted in the decay of ^{238}Np coming from ^{242m}Am alpha decay; for the ^{241}Am mass, the lines were used at 322.5, 332.4, 335.4, 376.6 and 722.0 keV emitted in the decay of ^{241}Am. A very long counting time: ~5E+5 sec was imposed for each sample (because of the continuous compton background) to get a statistical precision better than 10%.

The following results (in grams) have been found:

	Bare samples	**Cd-lined samples**	
^{241}Am	$(3.85 \pm 0.06)\times10^{-4}$	$(3.80 \pm 0.09)\times10^{-4}$	30 h irradiation
	$(2.99 \pm 0.71)\times10^{-4}$	$(4.60 \pm 0.46)\times10^{-4}$	119 h irradiation
^{242m}Am	$(5.56 \pm 0.08)\times10^{-8}$	$(1.16 \pm 0.27)\times10^{-8}$	30 h irradiation
	$(1.93 \pm 0.05)\times10^{-7}$	$(5.37 \pm 0.16)\times10^{-8}$	119 h irradiation,

Based on a ^{242m}Am build-up model, the expected values (both ^{241}Am and ^{242m}Am) are in good agreement with the experimental ones for the bare samples, while not-negligible discrepancies are observed for the Cd-lined samples. Several reasons, singly or concomitants, can contribute to such a behaviour.

Regarding the uncertainties of the experimental results, the uncertainties of ^{241}Am and ^{242m}Am behave in opposite way. An increase in the irradiation time can result an increase in the uncertainties (of several times) of ^{241}Am, as well as a decrease in the uncertainties of the ^{242m}Am. The behaviour that could be due to the increase in the activation of the other components of the samples, contributing to "dirty" the signal of the detecting gamma rays. A supplementary investigation for such a behaviour would be required.

Although experimental feasibility and validation of the ^{242m}Am build-up model have been demonstrated, the observed discrepancies and "inconsistencies", as well as the updating of the nuclear data suggest there should be a "re-evaluation" of the experiment.

3.4 Japan

In Japan, to validate and improve the reliability of nuclear data of minor actinides (MA), the basic integral experiments, such as measurements of fission reaction rate ratio and small sample reactivity worth were performed using the critical assemblies with fast and thermal neutron spectrum fields. Moreover, MA sample irradiation tests were carried out at JOYO, in the experimental fast reactor in Japan. In addition to the experimental results at JOYO, the sample irradiation results were also used for the validation and the improvement of nuclear data in PFR, which is the Proto-type Fast Reactor in the UK.

In the Fast Critical Assembly (FCA), experimental studies were carried out on fission rate ratios and sample reactivity worth for MA in various types of experimental configuration. Especially, in the FCA IX series experiments, in which seven different cores systematically changed neutron spectra were built, the measurements were carried out for the fission rate ratios and the sample reactivity worth for MA samples [38]. For the fission rate ratios measurements, fission chambers of parallel-plate type were used. The chamber consists of the body made of thin stainless steel wall, the circular disk-shaped anode and electrode and the fissile deposit of 25 mm diameter, which is electro-deposited on a polished platinum foil of 0.2 mm thick and 36 mm diameter, as shown in Figure 24. The gas filled in the chamber is a mixture of 97% Ar and 3% N_2 at 1 atmosphere. For most of the deposits, the fissile masses were adjusted in the range of 40~120 μg. However, for isotopes with very high α-specific activities such as ^{244}Cm, ^{238}Pu and ^{241}Am, the masses of deposits were limited to less than 10 μg, to reduce α pulse pile-up. The fissile masses in all deposits were determined by α-spectrometry using silicon surface barrier detector mounted in a vacuum chamber. The deposit mass was determined within ±1.5% error (±3.0% for ^{244}Cm). The composition of principal isotope is 100%, 78.4%, 96.7%, 99.8%, 100%, 99.6% and 79.0% for the ^{237}Np, ^{238}Pu, ^{239}Pu, ^{242}Pu, ^{241}Am, ^{243}Am and ^{244}Cm chambers, respectively. The actinide samples used in integral experiments on minor actinides at FCA were 15-20 grams of MA (^{237}Np, ^{238}Pu, ^{240}Pu, ^{241}Am, ^{243}Am) oxide powder that is packed in double cylindrical capsules of stainless steel. Most of the inside of the inner capsule is filled with the actinide oxide powder and stainless-steel spacers. Figure 25 shows that both ends of this capsule are sealed with stainless steel end caps. These samples were prepared by the Oak Ridge National Laboratory in 1980.

Figure 24: Parallel-plate type fission chamber

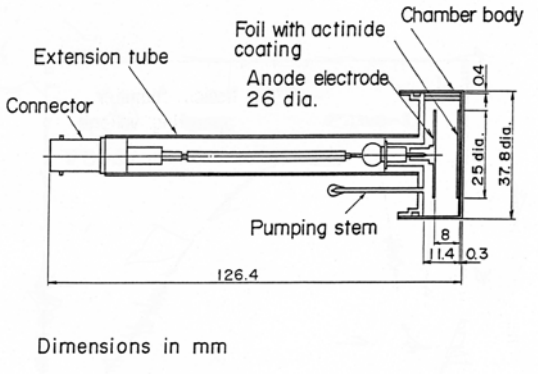
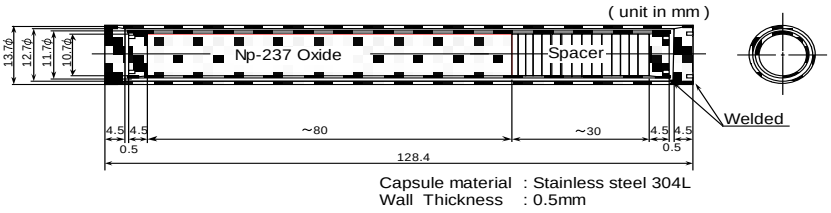


Figure 25: Cross-sectional views of ^{237}Np sample and aluminum container



In the Tank-Type Critical Assembly (TCA) of the JAEA, the sample reactivity worth of ^{237}Np and ^{241}Am was measured in seven different cores with low-enriched uranium fuel and light water moderator. These cores were built using fuel lattices with water-to-fuel volume ratios ranging from 3.00 to 0.56 so as to provide a systematic variation in the neutron spectrum between the thermal and resonance energy regions, so that a benchmark data base could be produced for the reactivity worth of ^{237}Np and ^{241}Am that were sensitive to the neutron spectrum in these energy regions. The ^{237}Np and ^{241}Am samples used for the experiments in TCA are the same ones used at the FCA.

In the Kyoto University Critical Assembly (KUCA), a series of integral experiments was performed for ^{237}Np and ^{241}Am fission rate ratio relative to ^{235}U fission rate [40-42]. The neutron spectrum of the experimental field was varied so as to systematically obtain useful integral data for the assessment of evaluated cross-section. Experiments were performed using polyethylene-moderated and -reflected, highly-enriched uranium cores constructed in a solid moderated assembly. In order to systematically vary the neutron spectrum of the experimental neutron field, the H/^{235}U ratio of the core was systematically varied by changing the combination of the polyethylene moderator plates and 93%-enriched uranium fuel plates in a unit cell. The fission rate ratio measurements were performed using the back-to-back type double fission chamber. Each sample is electrodeposited on stainless steel backing plate of 28 mm in diameter and 0.2 mm in thickness. The diameter of the deposit is 20 mm.

Separated actinide samples were irradiated at the Dounreay Prototype Fast Reactor (PFR) [43, 44]. The experiments were conducted under a joint research programme between the United States (US) and the United Kingdom (UK). The actinide samples were prepared by the Oak Ridge National Laboratory (ORNL). The samples were milligram quantities of actinide oxides of $^{230, 232}$Th, ^{231}Pa, $^{233, 234, 235, 236, 238}$U, ^{237}Np, $^{238, 239, 240, 241, 242, 244}$Pu, $^{241, 243}$Am, $^{243, 244, 246, 248}$Cm. These oxide powders were encapsulated in vanadium holders and loaded to the special irradiation pin, FP-4 (35 capsules consisting 9 dosimeter capsules and 26 actinide physics capsules). The pins were placed in two specially built clusters of 19 pins. The other pins in these clusters were standard fuel pins. In turn, six of these 19-pin clusters formed a demountable subassembly (DMSA). The arrangement of the cluster in a DMSA and the location of the DMSAs in the core are shown in Figure 26. The cluster containing FP-4 was irradiated from July 1982 to July 1988 at variable conditions with several periods of cooling outside of the core. At final discharge, the total irradiation for FP-4 amounted to 492 effective full-power days (EFPD). The total exposure and the total fluence were 295,483 MWd and ~2×1023 n/cm^2, respectively. The irradiated samples were dissolved and partitioned at ORNL. Parts of samples solutions were transported to JAERI (the former of JAEA) from ORNL. There was good agreement between the results of radiochemical analysis at JAEA and the results obtained at ORNL. The results of radiochemical analysis at JAEA were compared with the calculation results.

MA irradiation tests were performed at JOYO from 1994 to 1999 [45]. In the tests, MA samples of ^{237}Np, ^{241}Am, ^{243}Am and ^{244}Cm were loaded at the JOYO MK-II core. Each MA sample includes 100 µg of MA heavy metal. While the purity of ^{237}Np and ^{241}Am samples were almost 100%, ^{243}Am sample included ^{241}Am of 12% and ^{244}Cm sample included ^{245}Cm of 2% and ^{246}Cm of 4%, respectively. These samples had been irradiated in the core mid-

plane and upper reflector for 276 EFPD in the fuel irradiation test subassembly PFB090, as shown in Figure 27. After the MA irradiation tests, MA isotopic compositions were quantified by radiochemical analyses of MA samples of ^{241}Am, ^{243}Am and ^{244}Cm extracted from the subassembly PFB090. The estimated incineration fractions were approximately 10% for both Am samples and nearly 30% for ^{244}Cm samples.

Figure 26: Core layout of PFR showing DMSA experimental positions

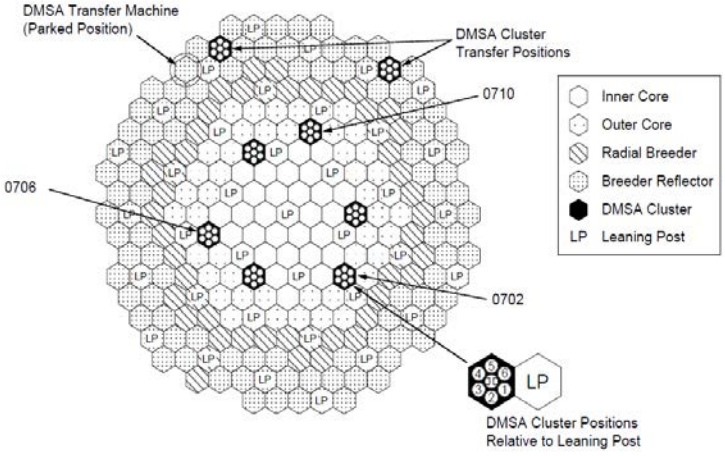
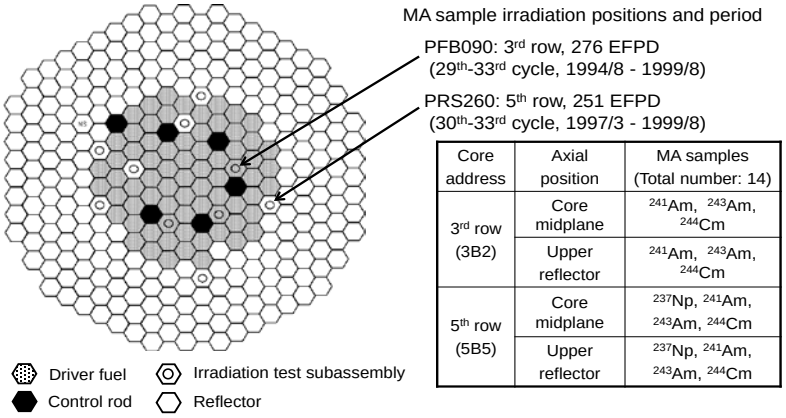


Figure 27: Core layout of JOYO MK-II (33rd cycle) and the possibility of the MA-loaded irradiation subassembly



Core address	Axial position	MA samples (Total number: 14)
3rd row (3B2)	Core midplane	^{241}Am, ^{243}Am, ^{244}Cm
	Upper reflector	^{241}Am, ^{243}Am, ^{244}Cm
5th row (5B5)	Core midplane	^{237}Np, ^{241}Am, ^{243}Am, ^{244}Cm
	Upper reflector	^{237}Np, ^{241}Am, ^{243}Am, ^{244}Cm

The accelerator-driven subcritical system (ADS) has been developed for transmuting MA. A series of preliminary experiments on an ADS with 14 MeV neutrons was conducted at KUCA with the prospect of establishing a new neutron source for research. A critical assembly of a solid-moderated and -reflected core was combined with a Cockcroft-Walton-type accelerator. A neutron shield and a beam duct were installed in the reflector region for directing as large a number as possible of the high-energy 14 MeV neutrons generated by deuteron-tritium (D-T) reactions to the fuel region, since the tritium target is located outside the core. Neutrons (14 MeV) were injected into a subcritical system through a polyethylene reflector. The basic experimental study is conducted to examine experimentally the neutronic properties of the ADS with 14 MeV neutrons.

3.5 Russian Federation

Basic experiments using critical facility

Facility: BFS-60, -71, -73*, -75*, -61**, -77**, -87** (IPPE, Russian Federation)

Starting from the 1990s, the wide experimental programme on MA constants investigations for different neutron spectrum was performed at the BFS critical assemblies [46, 47]. Measurements of fission cross-section ratios, central reactivity worths, and Doppler-effect were carried out. Experiments on the direct introduction of MA in the fuel composition and investigations of its influence on neutronic parameters such as sodium void reactivity effect, control rod worths and spectrum indexes were performed. Investigations of traditional cores with oxide and metal fuel and sodium coolant and testing cores with nitride fuel and Pb / Pb-Bi-coolant were carried out. The aim of the work is to create the total set of evaluated integral experiments on critical assemblies for the definition of the MA data.

* *Critical assemblies with sodium coolant and metal fuel.*
** *Critical assemblies with Pb and Pb-Bi coolant and nitride fuel simulation.*

Sample irradiation experiments

Facility: BN-350 (IPPE , Russian Federation)

At the beginning of the 1990s, a set of minor actinides, ^{241}Am, ^{237}Np, ^{238}Pu, ^{240}Pu, and ^{244}Cm, was irradiated in BN-350 fast power reactor [48]. The aim of the experimental work was to measure the ratio of the cross-sections of different nuclides in the nuclear fuel composition and structural materials and, the ratio of the material composition variations of different types of fuel and actinide samples under the influence of fast reactor neutron irradiation.

Fuel and actinide samples were irradiated at standard power level with further radiochemical analyses of the sample nuclide composition variation. These experiments are suitable for verification of fast reactor nuclear data and neutron calculation codes taking into account continuously varying nuclide composition.

Mock-up experiments

Facility: BFS (IPPE , Russia)

The models of fast reactors with sodium-coolant and MOX fuel were investigated in the critical assemblies BFS-62-2 and BFS-67 [47]. In these assemblies, the influence of the placement of all available NpO_2 (11 kg) in the form of BFS pellets on the main neutronic parameters was studied.

3.6 Switzerland

Swiss research reactors suitable for investigations of minor actinides are scarce. Investigations were performed in the PROTEUS zero power research reactor at the Paul Scherrer Institut. PROTEUS has been operated since 1968 and has allowed studying several reactor type concepts along the years, e.g. gas-cooled fast reactor (GCFR), high-conversion light-water reactor (HCLWR) and the high-temperature reactor (HTR). The large panel of reactor spectra is a significant asset to design experiments for minor actinides. Unfortunately, non-proprietary measurements involving minor actinides are rather scarce.

The main contribution to the database on minor actinides comes from the experimental programme on the gas-cooled fast reactor. Spectral indices such as fission in ^{237}Np (F7) over fission in ^{239}Pu (F9) and capture in ^{237}Np (C7) over fission in ^{239}Pu (F9) were

measured in two core configurations labelled “reference” and “moderated” [49]. In the reference configuration, the test zone of PROTEUS was loaded with a regular hexagonal lattice of PuO_2/UO_2 fuel rods containing 15% plutonium [50]. The median energy was about 185 keV. The moderated configuration had a softer spectrum and was similar to the reference core with additional polystyrene beds inserted between the lattice rods [51]. C7/F9 and F7/F9 were measured with 1σ uncertainty around 2%.

Results were compared with two 1-D deterministic calculations involving the ENDF/B-4 and UK FGL5 data set. The two data sets only significantly differ in their ^{237}Np capture cross-sections above 100 keV. More recently, results in the reference configuration have been compared with predictions coming from a full 3D MCNPX model that uses the JEFF-3.1 data library [52]. Table 20 shows the experimental results and C/Es. The F7/F9 spectral index is well predicted for all libraries and calculation routes, whereas the 3D MCNPX model coupled with the recent JEFF-3.1 data library correct the under-estimation of C7/F9 observed with the two older libraries. In summary, a modern calculation code with the modern library JEFF-3.1 adequately predicts the measured C7/F9 and F7/F9 spectral indices.

Table 20: Comparison of measured and calculated spectral indices in a GCFR-type spectrum

	Reference configuration		Moderated configuration	
	C7/F9	F7/F9	C7/F9	F7/F9
Experiments	0.826 (2.3%)	0.227 (1.8%)	0.957 (2.3%)	0.207 (1.9%)
C/E (FGL5)	1.102 (2.3%)	0.983 (1.8%)	1.111 (2.3%)	0.980 (1.9%)
C/E (ENDF/B-4)	0.881 (2.3%)	0.981 (1.8%)	0.950 (2.3%)	0.974 (1.9%)
C/E (JEFF-3.1)	0.949 (2.4%)	1.003 (1.8%)	-	-

3.7 United Kingdom

The zero energy breeder reactor assembly facility, ZEBRA, was used to study the neutron physics of fast reactors. It operated from 1962 until 1982 at the Winfrith site of the United Kingdom. Atomic Energy Authority Mock-up type assemblies were studied as well as simple assemblies for the validation of nuclear data and methods of calculation. Some experiments were designed to validate the methods used to model the heterogeneity of the composition of fast reactors.

Control rod studies and sodium voiding studies were carried out on several assemblies. Spectral indices, small sample reactivity worths and reaction rate distributions were measured in most assemblies. Neutron spectrum measurements were also made in several assemblies. Different techniques were used, including pulsed time of flight, proportional counters and solid state track recorders. In the later assemblies, a multi-chamber scanning system was used to measure ^{239}Pu fission rates simultaneously at a large number of points in the core.

During the 1970s, a series of measurements were performed to determine the cross-sections of many of the isotopes of uranium, plutonium americium and curium in a spectrum similar to that found in the UK Prototype Fast Reactor (PFR) which operated at Dounreay in Scotland. An overview of the measurements made during this period is shown in Tables 21 through 23. They include measurements of the fission cross-sections for ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{241}Am, ^{243}Am, and ^{244}Cm, as well as the capture cross-sections for ^{241}Am and ^{243}Am. Uncertainties were typically around 4% for the Pu and Am fission rates rising to just over 7% for the ^{244}Cm. Uncertainties on Am capture cross-sections were somewhat larger, in the range of 6-15% depending on nuclide and spectrum.

Table 21: Ratios of calculated fission rates (using FGL5 data) to experimental measurements in a fast power reactor spectrum (C/E) [53]

Nuclide	C/E
^{238}U	1.037±4.3%
^{240}Pu	1.003±4.7%
^{241}Pu	1.048±3.2%
^{242}Pu	1.231±5.0%
^{241}Am	1.260±4.1%
^{243}Am	0.882±4.1%
^{244}Cm	1.348±7.3%

Table 22: Comparison of experimental results with theory [54]

	Experimental value (E)	Calculated value (C)	C/E
σ	1.50±0.09 barns	0.97±0.06 barns	0.65
σ / σ_{239}	0.808±0.025	0.524±0.031	0.65

σ : microscopic cross-section

Table 23: Curium-production cross-sections (Am capture) [55]

Sample	Core	Fluence (n.cm^{-2})	Cross-section for Cm production (barns)		C/E
			Experiment (E)	Calculation (C)	
^{241}Am	14	2.97 x 10^{14}	1.28±0.08	1.12	0.88±0.06
	12	2.60 x 10^{15}	1.48±0.10	1.27	0.86±0.06
^{243}Am	14	2.66 x 10^{15}	1.32±0.20	1.58	1.20±0.18
	16	2.16 x 10^{15}	1.99±0.20	1.73	0.87±0.09

3.8 United States

An effort is on-going at INL/ANL/ISU to perform an integral reactor physics experiment whose objective is to infer actinide neutron capture cross-sections. The principle of this experiment, called MANTRA (Measurement of Actinide Neutron Transmutation Rates by Accelerator mass spectrometry), is to irradiate very pure samples in the Advanced Test Reactor (ATR) at INL and, after a given time, determine the amount of the different transmutation products. The determination of the nuclide densities before and after neutron irradiation will allow inference of energy-integrated neutron cross-sections.

This approach has been used in the past and the novelty of this experiment is that the atom densities of the different transmutation products will be determined using the Accelerator Mass Spectrometry (AMS) technique at the ATLAS Facility located at ANL. This technique is sensitive for measuring quantities of long-lived, rare isotopes with high

discrimination in the presence of more abundant ones using very small amounts of material. While AMS facilities traditionally have been limited to the assay of low-to-medium atomic mass materials, i.e. A < 100, there has been recent progress in extending AMS to heavier isotopes, even to A > 200. The detection limit of AMS is orders of magnitude lower than that of standard mass spectrometry techniques (abundances as low as 10^{-12} can be detected), thus allowing more transmutation products to be measured and consequently more neutron cross-sections to be inferred from a single sample.

It is currently planned to irradiate the following isotope samples: ^{232}Th, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{244}Cm and ^{248}Cm. These enriched isotopes are available at INL and their precise characterisation is underway. Only a few mg of each material will be needed in order to prepare the experimental samples. The ATR is a 250-MWth water-cooled reactor, with a thermal neutron spectrum. Using neutron filters (cadmium and boron) it is possible to modify the spectrum and thus meet specific needs. The cadmium and boron-filtered samples will be irradiated, respectively, for 55 days and 110 days. In April 2013, cadmium filtered samples were irradiated, and after cooling reached the facility used for dissolution. PIE results for these samples are already available and the irradiation of boron (thin and thick) filter samples was completed in 2013.

4. Discussion of experiments to meet the requirements

4.1 Summary of existing integral data

This section gives an overview about the exisiting integral experiments. The Expert Group members collated existing integral experiments related to minor actinide management. Table 24 summarises the integral reaction rate measurements in various experimental facilities. The integral reaction rates or reaction rate ratios have been one of the most basic experimental items. These integral data have been used in a variety of ways not only in the evaluation but also in the verification of neutron cross-sections. Moreover, these integral data have been frequently used for the adjustment of the multi-group neutron cross-section library. In the experiments, the reaction rates are measured using fission chamber and/or activation foils. Table 24 shows that there are relatively many numbers of integral data for the fission reaction rates measured in two or more experimental facilities. In the adjustment method, existence of many independent experimental data is important to exclude the systematic experimental error. However, there is no experimental data for ^{245}Cm which is one of the main contributors for the uncertainty. On the other hand, for the capture reaction, there are a few experimental results, especially in fast neutron field. It should be noted that difficulty of the sample preparation is the bottleneck, since a few tens of mg of samples with high purity will be needed for the capture reaction rate measurements.

Table 24: Summary of reaction rate measurement experiments

Fast neutron spectrum

	VENUS	MASURCA	FCA	BFS	GODIVA, JEZEBEL, FLATTOP
Mass	0.4-12 mg	Some 10 mg	40-120 mg	30-400 mg	Various
Purity	60 – 99 %	High	78-100 %	87 -99 %	High
^{232}Th		F			
^{234}U	F				
^{235}U		F			F
^{238}U		F			
^{237}Np	F	F	F	F, C	F
^{238}Pu		F	F	F	
^{239}Pu	F	F	F	F	F
^{240}Pu	F	F	F	F	
^{241}Pu		F		F	
^{242}Pu	F	F	F	F	
^{241}Am	F	F	F	F	C
^{243}Am		F	F	F	
^{244}Cm			F	F	

F: Fission, C: Capture.

Thermal neutron spectrum

	TRIGA RC-1	EOLE	PROTEUS*	KUCA
Mass	0.3-0.46 mg	Some 100 mg	Deposit	
Purity	100 %	~100 %		
^{232}Th				
^{234}U				
^{235}U				
^{238}U				
^{237}Np		F	F, C	F, C
^{238}Pu		F		
^{239}Pu		F		
^{240}Pu		F		
^{241}Pu		F	F	
^{242}Pu		F	C	
^{241}Am	C			F
^{243}Am				
^{244}Cm				

F: Fission, C: Capture

* Some experiments were carried out in a fast spectrum arranged in the central part of PROTEUS hybrid core.

Table 25 summarises the small sample reactivity measurements in various experimental facilities. Generally, the reactivity of a sample is measured by inserting it into and out of the core, while a calibrated control rod maintains a constant power level. In some cases, however, the measurements are performed by the reactor period method (open loop). From the difference in average position of the control rod between the sample-in and sample-out cases, the reactivity is determined. Although systematic measurements of OSMOSE programme have been carried out at MINERVE, there are few experimental results in the fast neutron spectrum field. Although the comparison with the experimental data and the calculation results has been effective for the verification of the neutron cross-section, careful consideration is needed as described below. The perturbation theory expressed the sample reactivity in the following equation:

$$\rho = \frac{\left\langle \phi^* \Delta\Sigma \phi \right\rangle}{IP} = \frac{\left\langle \phi^* \Delta\Sigma_f \phi \right\rangle}{IP} + \frac{\left\langle \phi^* \Delta\Sigma_c \phi \right\rangle}{IP} + \frac{\left\langle \phi^* \Delta\Sigma_s (E \to E') \phi \right\rangle}{IP}, \tag{11}$$

where denominator IP is the perturbation denominator which is importance-weighted neutron production rate. From this equation, it is clear that the reactivity depends on the fission term [+], the capture term [-] and the scattering term [+/-]. In some cases, the value may become near zero since the positive contribution and the negative contribution cancel each other out. Consequently, the C/E may become a greater value than unity. Moreover, the sensitivity coefficient of the sample reactivity consists of two terms; the sensitivity of the numerator term and that of the denominator term in the reactivity change.

$$S = \frac{\partial\rho / \rho}{\partial\Sigma / \Sigma} = \frac{\frac{\partial\rho_N}{\rho_N} - \frac{\partial\rho_D}{\rho_D}}{\frac{\partial\Sigma}{\Sigma}} \equiv S_N - S_D \quad \text{where} \quad \rho = \frac{\left\langle \phi^* \Delta\Sigma \phi \right\rangle}{IP} \equiv \frac{\rho_N}{\rho_D} \tag{12}$$

Therefore, since the sensitivity coefficient may be influenced by the core materials, the experiments eliminating or reducing the effect of the core material will be necessary.

Table 25: Summary of small sample reactivity measurement experiments

Fast neutron spectrum

	FCA	BFS	JEZEBEL	FLATTOP
Mass	15-20 g	15-20 g	3-12 g	~30g
Purity		99 %	80-95%	93-99%
^{232}Th				
^{234}U				
^{235}U			X	X
^{238}U				
^{237}Np	X	X		X
^{238}Pu	X		X	
^{239}Pu			X	
^{240}Pu	X			
^{241}Pu				
^{242}Pu				
^{241}Am	X	X		
^{243}Am	X			
^{244}Cm			X	

Thermal neutron spectrum

	MINERVE	TCA	KUCA
Mass	0.1-1.9 g	15-20 g	
Purity	100 %	99(^{237}Np), 90(^{241}Am)	
^{232}Th	X		
^{234}U			
^{235}U			
^{238}U			
^{237}Np	X	X	
^{238}Pu	X		
^{239}Pu	X		
^{240}Pu	X		
^{241}Pu	X		
^{242}Pu	X		
^{241}Am	X	X	X
^{243}Am	X		
^{244}Cm	X		

Table 26 summarises the sample irradiation experiments in various experimental reactors. Irradiation experiments are extremely useful for validating data related to minor actinides. The results of the analysis of such experimental data provide indications to nuclear data evaluators for improving the quality of basic files. Some experimental data, such as samples irradiated in PFR and PROFIL experiments in Phenix, provide very clean and useful information on both cross-section data and transmutation rates of actinides. These data are essential for the verification of the neutron cross-section in the evaluated nuclear data library. Moreover, in the validation process the use of sensitivity analysis allows indicating information and specific needs. However, there are few such experiments, and many were performed a long time ago. Some information necessary for the detailed analysis is still missing.

Table 26: Summary of sample irradiation experiments in fast and intermediate spectrum

	PFR	PHENIX PROFIL-1, 2	PHENIX PROFIL-R, M	PHENIX SUPERFACT	JOYO	BOR-60	BN-350	ATR
Mass	0.4-12 mg	10-20 mg	few mg		0.1-0.2 mg	110-420 g	4-150 g	~1 mg
Purity	60 – 99 %	High	High	High	88-100 %	High	65-100 %	High
Irradiation period [EFPD]	492	179, 316	252, 227	382	250-280			50-150
^{230}Th	X							
^{232}Th	X	X	X			X		X
^{233}U	X	X	X					
^{234}U	X	X	X				X	
^{235}U	X	X	X					X
^{236}U	X							
^{238}U	X	X	X					X
^{237}Np	X	X	X	X	X	X	X	X
^{238}Pu	X	X	X				X	X
^{239}Pu	X	X	X			X		X
^{240}Pu	X	X	X			X	X	X
^{241}Pu	X	X						X
^{242}Pu	X	X	X			X		X
^{244}Pu	X							X
^{241}Am	X	X	X	X	X	X	X	X
^{243}Am	X	X	X		X	X		X
^{243}Cm	X							
^{244}Cm	X	X	X		X	X	X	X
^{246}Cm	X							
^{248}Cm	X							X

In the development of nuclear reactors with un-experienced material, integral critical experiments with un-experienced material were carried out for the design and the licensing. In the case of a core with massive target material, the critical experiments loading a massive target material loading were performed to measure various reactor physics parameters, since the neutron field is strongly influenced by the target material. This is impossible by sample irradiation or sample reactivity measurements. In the past, many critical experiments with massive major actinide, U, Pu, and Th, were performed in various countries. However, for MA, there are only the experiments with large amount of ^{237}Np (about 10 kg of ^{237}Np-doxide pellets) performed in the BFS critical assembly in the IPPE, Russian Federation. The calculational study [56] using the cross-section adjustment method showed that clear reduction of the uncertainty after the cross-section adjustment was found, especially for sodium void reactivity, and the uncertainty reduction was caused by the important nuclides and reaction, such as ^{239}Pu ν and fission, which were strongly correlated after adjustment.

Many integral experiments have been largely performed for the neutronics studies of fast critical systems. On the other hand, for ADS, in which a subcritical reactor is driven by the external neutron source from the accelerator beam to maintain the chain reaction, some experimental studies have been carried out in some countries. In Europe, for example, there have been MUSE experiments at MASRUCA Facility and GUINEVERE experiments at VENUS Facility. In these experiments, in order to validate the reactor physics characteristics of the subcritical system driven by external neutron source, well-known neutron source (14 MeV neutron source via DT reaction) was used to separate the validation of the external neutron source. Major items to be investigated by these experiments were the validation of the subcriticality monitoring for an ADS and applicability of calculation system developed for the fast critical reactor. In Japan, ADS experiments have been conducted in a well-thermalised critical assembly (KUCA A-core) in order to evaluate the basic characteristics of ADS by coupling with 14 MeV (D,T) accelerator and spallation neutron generated by FFAG proton accelerator.

4.2 Specification of missing integral data

The Expert Group members gathered the existing integral experiments, especially related to the MA management. This section describes the collected integral experiments.

Reaction rate

As described in the previous section, there are relatively many integral data for the fission reaction rates measured in two or more experimental facilities. It is desirable to continue the measurements of the reaction rate measurements. However, since there is no experimental data for ^{245}Cm, which is one of the main contributors for the uncertainty, measurement for fission reaction rate of ^{245}Cm is suggested. On the other hand, for the capture reaction, there are only few experimental results, especially in the fast neutron field. Therefore, more measurements would be desirable.

Sample reactivity

Although systematic measurements have been carried out at MINERVE as part of the OSMOSE programme, there are very few experimental results in the fast neutron spectrum field. Along with the extension of the integral experimental data in the careful experimental condition, adequate usage method of the experimental results for the cross-section adjustment should be considered.

Sample irradiation

The sample irradiation experiments are essential for the verification of the neutron cross-section in the evaluated nuclear data library. However, many irradiation experiments were performed a long time ago, some information which is necessary for the detailed analysis is still missing. The sample irradiation experiments will take a long time for the sample preparation, irradiation and post-irradiation experiment. The verification of the accessibility to the existing data and the availability of the existing data are necessary for further analytical studies.

Mock-up experiment

In the development of the nuclear reactor with un-experienced material, integral critical experiments with un-experienced material were carried out. However, for MA, there are only the experiments with large amount of ^{237}Np (about 10 kg of ^{237}Np-doxide pellets) performed in the BFS critical assembly in the IPPE, Russian Federation. For other MA, such as Am, the possibility of the critical experiment with massive MA should be considered.

For ADS, some experiments with 14 MeV (D,T) accelerator have been performed. Experiments with the spallation neutron source have been conducted only at KUCA. Combined experiments with fast neutron subcritical core and the spallation neutron source should be considered.

4.3 Identification of bottlenecks and possible solutions

Sections 2 and 3 present the important nuclides of MA for design of transmutation systems and for fuel cycles: ^{237}Np, ^{241}Am, ^{242m}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm and ^{245}Cm. To improve the cross-section data for them and to design the MA transmutation system with reliable accuracy, many additional integral data are still necessary. The Expert Group members discussed the availabilities of minor actinides and experimental facilities to identify bottlenecks and to find solutions.

4.3.1 Availability of minor actinides

To carry out additional experiments, adequate amounts of minor actinides with high purity should be prepared. As shown in the previous section, a few hundred of mg or less of minor actinides are needed for both reaction rate ratio measurements and irradiation experiments, a few tens of grams of the materials are needed for the small sample reactivity worth measurements and several tens of kg of the materials are needed for the mock-up experiments and the criticality measurements. The members discussed the availabilities of minor actinides.

The Research Institute of Atomic Reactors (RIAR) in the Russian Federation has six research reactors and some complexes for post-irradiation examinations of irradiated materials and for investigation and production of transuranic elements and various radioisotope products. The institute has produced routinely radioisotopes and has sold them. RIAR are expected to supply the following materials of a few tens of grams or less on a commercial basis; ^{237}Np, ^{241}Am, ^{243}Am, ^{244}Cm, ^{245}Cm, ^{247}Cm, ^{248}Cm, ^{238}Pu, ^{240}Pu, ^{242}Pu. The CACAO group, Chimie des Actinides et Cibles radio Actives à Orsay (actinide chemistry and radioactive targets at Orsay) in France, will also supply the materials. The CACAO group was established as a project which aims to be a joint CNRS-CEA national laboratory in France. The group consists of the installation of a hot laboratory dedicated to the production and characterisation of thin radioactive layers. The group will be able to produce and/or characterise targets; ^{237}Np, ^{243}Am, ^{245}Cm, ^{247}Cm, ^{248}Cm, ^{240}Pu, ^{242}Pu, ^{244}Pu. In the US, the materials, ^{241}Am, ^{243}Am, ^{244}Cm, ^{248}Cm, ^{248}Cm, ^{238}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, are commercially available with sales by NIDC (National Isotope Development Centre).

On the other hand, for the mock-up experiments and the criticality measurements, it is difficult to procure massive amounts of MA. Prior starting the experiments, the selection of MA and the estimation of their amounts should be considered from the viewpoints of the integral experiments. Problems and solutions should also be considered in handling minor actinides during experiments, storage and transportation from the fabrication site to the experimental site.

4.3.2 Availability of experimental facility

The members surveyed the status of the experimental facilities. For the critical facilities which can be used for the basic experiments, the facilities for fast system, BFS (Russian Federation), FCA (Japan), MASURCA (France) and VENUS-F (Belgium) are available in the whole world.

For the irradiation experiments, three reactors, BOR-60 (Russian Federation), JOYO (Japan) for fast system and ATR (USA) for thermal system, are available. JHR (France) will start operation in the near future and MONJU (Japan) will be able to be used for the irradiation experiments in the near future.

For the ADS experiments, the thermal neutron water-moderated facility VENUS (Belgium) was modified into a fast neutron facility VENUS-F with solid lead core components in the period 2007-2010. The deuteron accelerator, so-called GENEPI-3C, was also installed to start to develop lead-based fast systems and R&D activities in support of the accelerator-driven systems. The experimental project, GUINEVERE, started in 2010. The GUINEVERE project flew into the FREYA project which started in March 2011. At another thermal neutron critical facility KUCA (Japan), the construction of FFAG proton accelerator was started to generate spallation neutron sources. The first neutron beam from the FFAG system was injected into a thermal subcritical system in 2009.

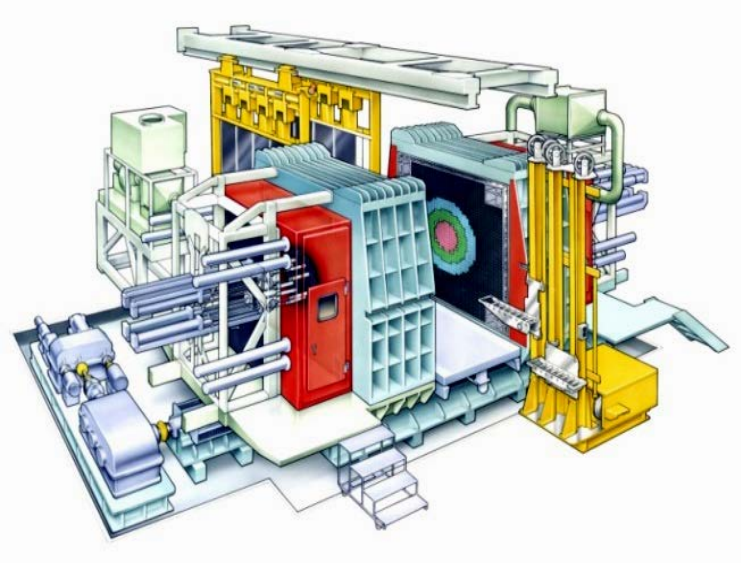
This section describes each available facility below:

4.3.2.1 FCA

The FCA fast critical facility in Japan was designed and constructed for studying physics characteristics of fast breeder reactor cores, improvements of data and method in fast and intermediate neutron energy range and developing measurement techniques. The first criticality was achieved in 1967. Since then, a lot of experiments have been carried out to provide integral data not only for core design of fast breeder reactors but also for the evaluation of nuclear data by building various experimental cores.

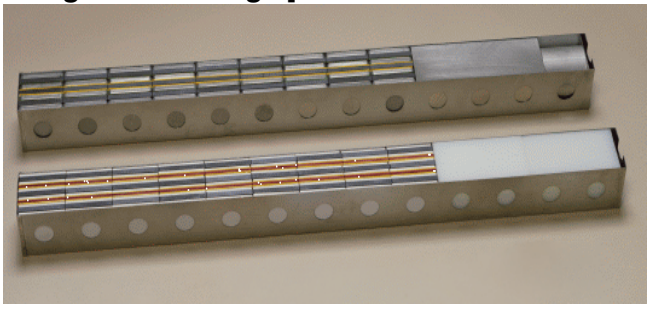
FCA is a horizontal split-table type machine consisting of a cast-steel bed supporting two tables, one stationary and the other movable. A bird's eye view of the FCA Facility is shown in Figure 28. Each table was 3-m wide and 1.3-m long. During loading operations, the tables were separated by 2 m. In operation, the movable table was driven at three different forward speeds against the stationary table. Stainless steel square tubes are stacked horizontally on both tables to form a 51-row and 51-column square "honeycomb" matrix. The (horizontal) control rods projected through holes in a steel back plate, and the drive mechanisms were positioned on the outside of this back plate. Each half assembly contains either four or five dive mechanisms for control/safety dual purpose rods of fuel removal type. One or two drawers are mounted with each drive mechanism as a rod and one of the rods in each half is used as control rod, and the others as safety rods.

Figure 28: Bird's eye view of FCA



The desired average composition was achieved by loading the matrix with drawers containing rectangular plates (or, in some assemblies, with cylindrical rods) of different materials such as depleted, enriched, or natural uranium; stainless steel; sodium, etc. The plates were bare material or had a cladding or a protective coating. Figure 29 is a photograph of the drawers forming a unit cell with using the plates for a particular region in a particular loading. The specification of which drawer was in each matrix position is known as a matrix loading map. The FCA drawers themselves typically are made of approximately 0.8-mm-thick stainless steel, and their front, back, and side walls are 51 mm tall.

Figure 29: Photograph of the drawers in FCA

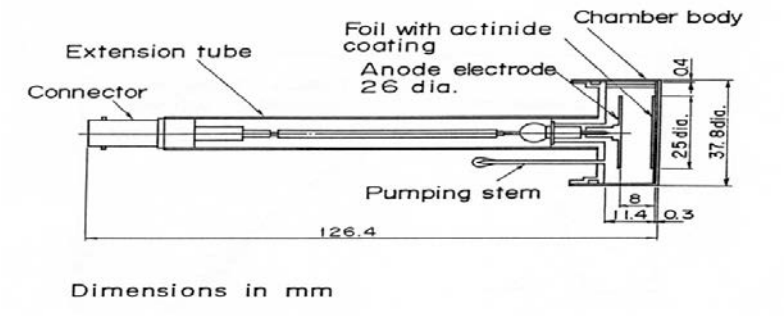


At FCA, the core material composition and the core geometry can be easily changed. Consequently, FCA has the following marks: 1) large flexibility in composition, 2) large flexibility in geometry and 3) great variety of neutron spectrum achieved. In taking advantage of these marks, the various experimental studies have been carried out at FCA. In these studies, various reactor physics parameters of the experimental cores were measured such as criticality, central reaction rate ratios, central small sample reactivity worths, reactivity effect caused by materials removed/replacement, etc. One of the typical experimental studies was FCA IX series experiment, as shown in Sections 3 and 4, whose purpose was to evaluate the minor actinide cross-section data [57-59].

Eight different cores in the series were built to change spectrum shift systematically. The integral data measured in these circumstances can provide more systematic information about neutron energy dependency than the data obtained in the reactors. Each core is composed of a core region and axial and radial blanket regions.

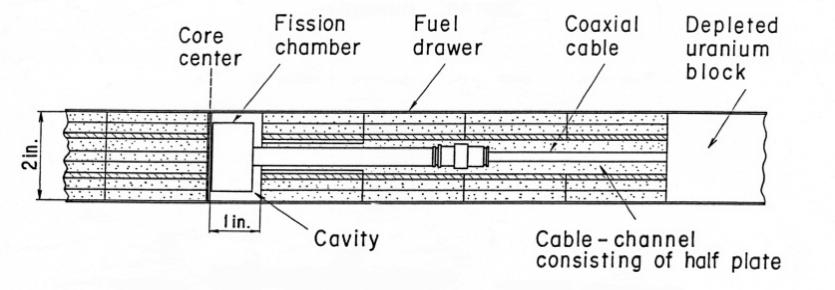
At FCA, fission chambers of parallel-plate type [60] are available to measure the fission rate ratios for minor actinides. As shown in Figure 30, the chamber consists of the body made of thin stainless steel wall, the anode electrode of a circular disk and the fissile deposit of 25 mm diameter which is electro-deposited on a polished platinum foil of 0.2 mm thick and 36 mm diameter. The gas filled in the chamber is a mixture of 97% Ar and 3% N_2 at 1 atmosphere. For most of the deposits, the fissile masses are adjusted in the range of 40~120 μg. However, for isotopes with very high α-specific activities such as ^{244}Cm, ^{238}Pu and ^{241}Am, the masses of deposits are limited to less than 10 μg, to reduce α pulse pile-up. The fissile masses in all deposits are determined by α-spectrometry using silicon surface barrier detector mounted in a vacuum chamber. The deposit mass is determined within ±1.5% error (±3.0% for ^{244}Cm).

Figure 30: Fission chamber of parallel-plate type



The fission chambers are mounted in a small cavity provided in the central FCA fuel drawer and inserted into the core centre. Figure 31 shows the location of fission chamber mounted in the drawer as a typical example. Dimensions of the cavity are 2"×2"×1". The conventional electronic system was used for measurement.

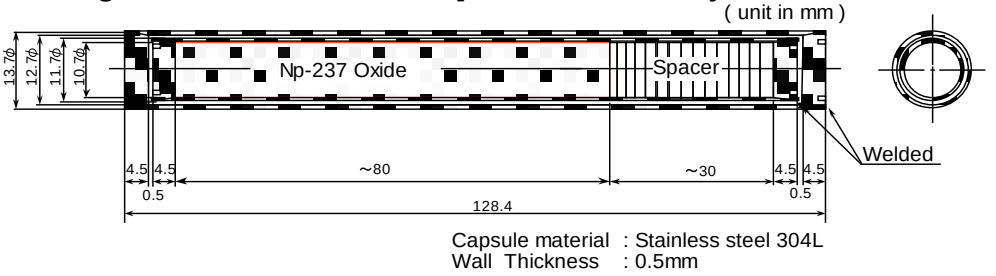
Figure 31: Fission chamber mounted in the central FCA drawer



During the fission rate measurements, reactor power level is monitored by two conventional fission chambers. The reactor is normally operated in 10~40 W power in response to sensitivities of the fission chambers, and fission pulses of 1.5×10^4~1.0×10^5 counts are accumulated.

For the sample reactivity worth measurements, the cylindrical samples of isolated actinide nuclides of ^{237}Np, ^{238}Pu, ^{240}Pu and ^{241}Am are available. The sample materials, usually about 15 grams in weight for Pu samples and 20 grams for the other samples, are loosely packed oxide powders sealed in double cylindrical stainless-steel capsules. Figure 32 shows the structure and the dimension of the capsule.

Figure 32: Minor actinide sample for FCA reactivity measurement



The samples are introduced into the core centre through the stainless-steel tube which penetrates radially the experimental core. The radial channel is created to reeve the tube by using plates containing holes. The wall thickness of the tube is 0.5 mm and the inner diameter is 17 mm. The tube was installed along a 25.4-mm radius from the axial mid-plane.

The reactivity of the perturbation introduced by a sample is measured by the technique that involves oscillation of a sample to eliminate the reactivity drift problem caused by temperature drift or the very small position change (order of a few tens of microns) of the movable side of the half assembly even after two halves of the assembly are closed.

The calibrated auto control-rod holds the reactor power constant while a sample oscillated. The auto control-rod position is recorded repeatedly when a sample is outside the assembly (the reference status) while a sample is oscillated. The change of the auto control-rod position provides the reactivity drift in time. The sample reactivity relative to the reference status is then related to the auto control-rod position change between a sample placed at the location of interest and the interpolated reference status at the same time.

The same measurements are repeated for the empty capsule. The reactivity difference between of the sample and of the empty capsule both relative to the reference gives the reactivity of the content, namely actinide powder.

4.3.2.2 MASURCA and MINERVE

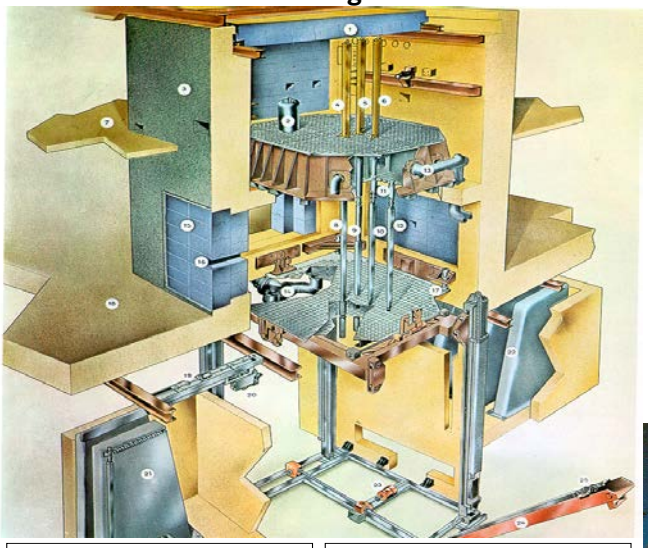
- MASURCA

MASURCA is a "zero power" critical facility operated by the CEA at the Cadarache Research Centre, France. It was designed for studying physics of fast spectrum lattices. The facility started in 1966. Various programmes were conducted without any interruption until 2006. Their results have been largely used for nuclear data library improvement, calculation code system validation and experimental technique development. The MASURCA Facility is now engaged in a refurbishment programme. The re-start of the experimental activity was planned for 2013.

A MASURCA core consists of an arrangement of subassemblies, having a square section (~10.5 cm x 10.5 cm) and a total length of about 3.8 metres, suspended to an upper plate structure that allows building cores with a diameter greater than 3 metres, as shown in Figure 33. These subassemblies, called "tubes", are made of a stainless steel wrapper containing a mixture of small fissile and non fissile elements, rodlets and platelets of various dimensions, which are assembled according to specific requirements.

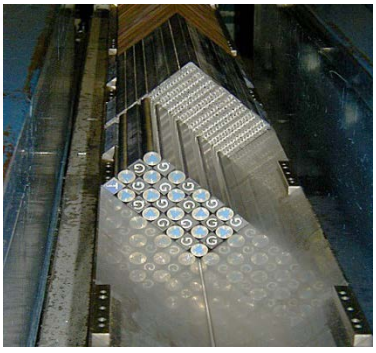
A large variety of nuclear materials available at MASURCA include oxide of thorium, metallic uranium with up to 35% of ^{235}U, oxides of uranium (natural, depleted and enriched with 30% of ^{235}U) and several plutonium fuels with various ^{240}Pu contents (8 → 44%): oxide of plutonium, mixed oxides of plutonium and uranium, metallic plutonium. Large amounts of inert materials – metallic sodium, Fe_2O_3, stainless steel, graphite, lead, void rodlets, etc. – are also available to simulate the coolant, the blankets, the reflectors as well as the structure materials.

Figure 33: MASURCA Facility



1: Moving slab
2: Starting source
3: Biological shielding
4, 5: Safety rod mechanisms
6: Pilot rod mechanism
7: Level +4,94 m
8, 9: Safety rods
10: Pilot rod
11: Suspension nozzle
12: Tube
13, 14: Ventilation circuit

15: Removable biological shielding
16: Radial measurement channel
17: Centering plate
18: Level +0,00 m
19, 20: XY handling carriage
21: East vertical storage
22: West vertical storage (not built)
23: Centering plate mechanism
24: Tipper
25: Tube head



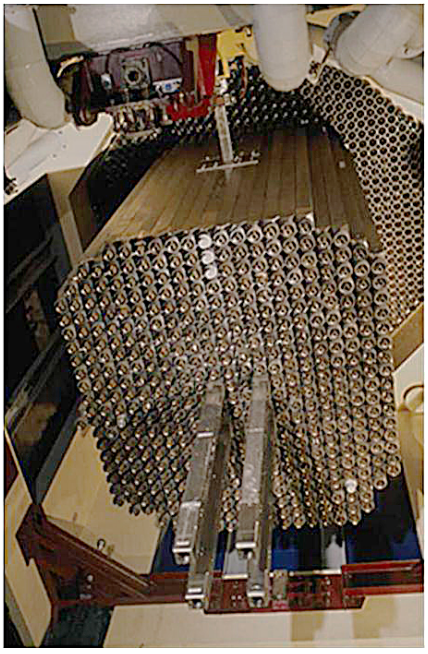
For safety purposes, the core also includes absorber rods; their number and position depend on the core type and size. To make the core critical and adjust the power, a pilot rod with a reactivity worth of less than half a delayed neutron fraction is used. The safety and pilot are designed in order to avoid perturbations on the measurements.

To perform local reaction rate measurements, axial channels can be built in every tube in which different devices can be introduced such as miniature fission chambers, fissile or inert activation foils, neutron sources. Dosimeters can also be inserted between or inside the rodlets. In addition, two horizontal radial channels (13x13 mm), at 90° of each other, can be set up near the mid-plane. Larger channels can be opened if necessary.

Basic experimental set-up consists of a) devices for moving the detectors or neutron sources inside the radial and axial channels, b) a large range of calibrated fission chambers (from ^{232}Th to ^{244}Cm) and commercial detectors, c) and modern acquisition systems. Moreover, a dosimetry laboratory is located in the vicinity of the MASURCA facility.

A large variety of core lattices and configurations have been investigated at MASURCA since 1966 (Figure 34). Until the early 1990s, the programmes (RACINE, BALZAC, CONRAD) were essentially dedicated to the development of sodium-cooled fast reactors such as PHENIX, SUPERPHENIX, EFR.

Figure 34: MASURCA core



Recent programmes were performed in the framework of the 1991 French Radioactive Waste Management Act. The CIRANO programme, which started in 1994, aimed to study plutonium-burning cores.

In the late 1990s, the COSMO programme was launched to study the transmutation of long-lived fission products targets in moderated subassemblies in fast neutron reactors. In the first phases of the programme, a moderated subassembly, consisting of a sodium zone surrounded by 11B4C pins and a steel layer, was placed at the core centre.

As part of the studies on the transmutation of long-lived nuclear waste in accelerator-driven systems (ADS), the MUSE programme was started with the objective of studying the physics of such a concept and the key issue of reactivity control. The initial MUSE configurations (MUSE-1, MUSE-2, MUSE-3) in the late 1990s were rather short test experiments. The more extensive MUSE-4 experiments were carried out from 2000 to 2004 and partially funded by the European Commission as part of the fifth Euratom Framework Programme. A reference critical core and several subcritical configurations with reactivity levels ranging from near-criticality to –14$ (~-4500 pcm) were investigated. In the very last phase of the programme, a configuration where sodium was partially replaced by lead was studied. Measurements were made using different external sources such as the core intrinsic source, ^{252}Cf sources, and 2.7 or 14 MeV neutron sources provided by a specially-designed neutron generator called GENEPI. A large number of experimental techniques were used such as the traditional rod drop/MSM method, source-jerk methods, variations of the pulsed neutron source methods, and noise methods.

Many of the past MASURCA programmes were performed under international co-operation. Experimental benchmarks were performed to compare different experimental techniques and analysis methods with other laboratories, such as control rod worth measurements, spectral indices (IRMA benchmark – 1984) and delayed neutron fraction (BERENICE programme in 1993 and 1994).

MASURCA is currently unavailable as it is undergoing a major refurbishment. The facility is expected to restart in 2017.

- MINERVE

The MINERVE reactor (Figures 35, 36) is made of a 70 cmx70 cm central experimental zone in which various material lattices can be loaded so as to create different spectral conditions, surrounded by a light-water-cooled driver zone made of highly-enriched metallic uranium plates. It is equipped with a special central tube in which small samples of reactor materials can be inserted and removed from the core in a repeated, periodic manner by a special driving mechanism. The reactivity variation induced by these oscillations is compensated by a pre-calibrated pilot rod mechanism in closed-loop fashion, so that the reactor remains critical. The measured reactivity variations are very accurate. They can be conveniently related to the main isotope cross-sections in the sample.

The reactor achieved its first criticality in 1959 at the centre of Fontenay-aux-Roses (near Paris) and was transferred to Cadarache in 1977. The core is built in a parallelepipedic pool of stainless steel of about 140 m^3, cooled by natural convection. The reactor is controlled using four hafnium rods. The thermal flux is about 109 $n/cm^2/s$ (100 Watts).

The available materials at Cadarache make it possible to simulate a broad range of reactor lattices, including light-water-cooled UOX and MOX lattices, under-moderated lattices, and even fast-spectrum lattices (ERMINE). Many different separated isotope samples are available, including actinides, fission products, absorbers and structural materials. Fuel samples fabricated from fuel rod segments irradiated in commercial reactors at different burn-ups are also available.

An example of a recent experiment is the OSMOSE programme, which started in 2005 and was completed in 2012. This programme consisted of actinide oscillations in several spectra. The actinides included ^{232}Th, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{244}Cm, and ^{245}Cm.

Figure 35: MINERVE reactor

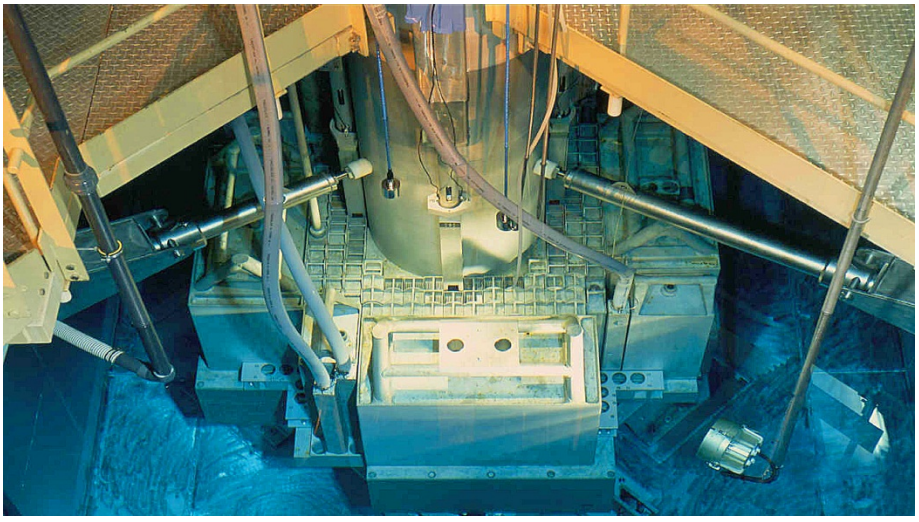
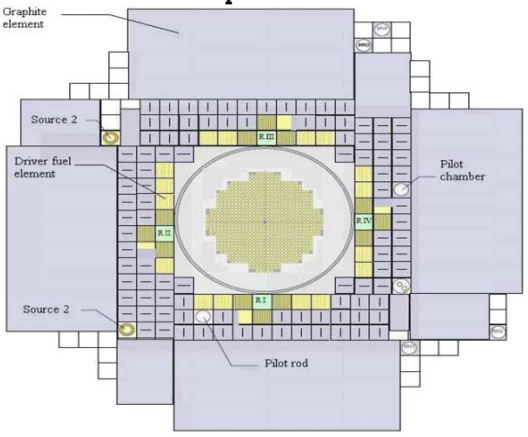


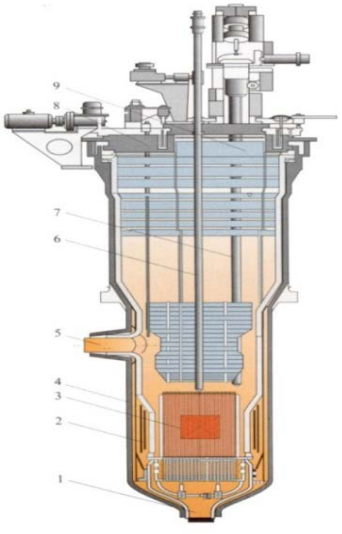
Figure 36: MINERVE central experimental zone and the driver zone



4.3.2.3 BOR-60

The BOR-60 Facility [61] is a fast, Na-cooled reactor operating on either 90% enriched uranium oxide or mixtures of up to 40% plutonium oxide. The facility started operation in 1969 and was designed to test fuel elements, structural materials, and fast-reactor safety issues for developments of advanced fast, pressurised water, gas-cooled and fusion reactors and to serve for substantiation of the VVER and BN-type reactor service life extension. The cylindrical core, which measures 40.4 cm in diameter by 40 cm high, is surrounded by a stainless steel reflector and a depleted uranium-oxide breeder region, as shown in Figure 37. The hexagonal assemblies have a minimum width of 4.4 cm. Table 27 shows the main characteristics of BOR-60. BOR-60 can operate at 60 MW. Nominal operating temperatures are in the range from 310 to 530^{0}C. Some positions are temperature controlled and are instrumented for materials experiments. The maximum neutron flux density is 3.7x1015 n/m^2-s and BOR-60 is available for suitable radiation experiments.

Figure 37: BOR-60



1: Inlet, 2: Reactor vessel shielding, 3: Core, 4: Reactor vessel, 5: Outlet, 6: Control rod, 7: Reloading channel, 8: Big rotating closure plugs, 9: Small rotating closure plugs.

Table 27: Main characteristics of BOR-60

Reactor heat power	60 MW
Maximum neutron flux density	3,7 10^{15} $sm^{-2}s^{-1}$
Maximum specific power	1100 kW/l
Average core neutron energy	0,45 MeV
Fuel	UO_2, UO_2-PuO_2
^{235}U enrichment	45÷90 %
Fuel burn-up rate	up to 6% per year
Neutron fluence per year	$5 \cdot 10^{22}$ sm^{-2}
Damage dose rate	up to 25 dpa /year
Inlet temperature of coolant	310÷330°C
Outlet temperature of coolant	up to 530 °C
Microcompaign duration	up to 120 days
Reactor generates energy	~265 days per year
Cells quantity - for S/A - for absorbing rods - instrumented cells	265 156 7 3

Figure 38: Cross-section view of the experimental assembly installed two pins (red ones) including eight quartz capsules containing pure actinides in BOR-60

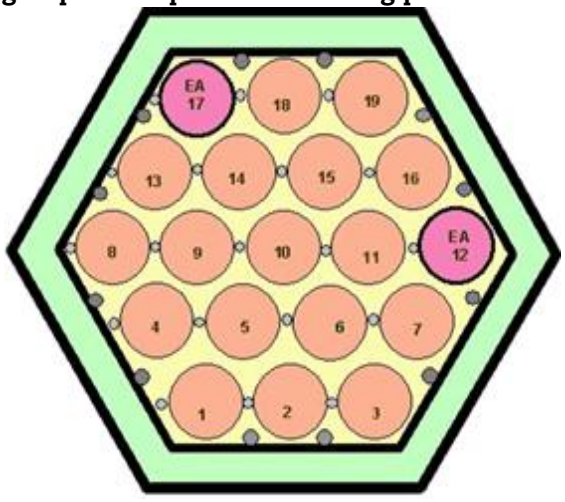


Table 28: MA isotope composition and samples burn-up after irradiation in BOR-60

Capsule No	Principal isotope	Isotope composition			Total burn-up (%)
		Isotope	Mass (micro gram)		
			Initial	After irradiation	
1	^{232}Th	^{232}Th	419 (100%)	410.6	0.26
		^{231}Pa		0.051	
		^{233}Pa		0.70	
		^{233}U		6.5	
		^{234}U		0.045	
2	^{237}Np	^{234}U	408 (100%)	0.26	4.7
		^{237}Np		360.2	
		^{238}Pu		27.6	
		^{239}Pu		0.39	
3	^{239}Pu	^{238}Pu	0.414 (0.36%)	0.36	13.8
		^{239}Pu	109.02 (94.8%)	90.8	
		^{240}Pu	5.35 (4.66%)	7.3	
		^{241}Pu	0.207 (0.18%)	0.57	
		^{242}Pu	0.0115 (<0.01%)	0.019	
		^{241}Am		0.042	
4	^{240}Pu	^{238}Pu	0.227 (0.18%)	0.19	5.8
		^{239}Pu	1.26 (1.0%)	1.05	
		^{240}Pu	124.51 (98.82%)	109.0	
		^{241}Pu		7.9	
		^{242}Pu		0.098	
		^{241}Am		0.42	
5	^{242}Pu	^{242}Pu	126.42 (99.54%)	117.2	3.8
		^{243}Am		4.3	
		^{244}Cm		0.13	
6	^{241}Am	^{234}U	129 (100%)	0.049	4.7
		^{237}Np		0.50	
		^{238}Pu		5.7	
		^{239}Pu		0.08	
		^{242}Pu		1.6	
		^{241}Am		110.5	
		^{242m}Am		1.7	
		^{243}Am		0.05	
		^{242}Cm		2.6	
7	^{243}Am	^{238}Pu	2.678 (2.37%)	0.11	3.3
		^{241}Am		2.32	
		^{243}Am	110.31 (97.62%)	100.9	
		^{244}Cm	0.0113 (<0.01%)	5.48	
		^{245}Cm		0.084	
8	^{244}Cm	^{244}Cm	129 (100%)		4.5
		^{245}Cm			
		^{246}Cm			

There are four complexes close to BOR-60 in RIARR: 1) Material Science Complex for full cycle of PIE for all type of Russian industrial Nuclear Power Plants' Fuel Analyses, 2) Radiochemical Complex for production of different type of isotopes including market scale batch of Am, Cm, Cf, 3) Radiochemical Complex for advanced fuel cycle development based on pyro technology of fuel production and reprocessing and vi-pack technology of fuel pins production and 4) Technology Complex for RAW Treatment included the unique super-deep underground Polygon of LLW disposal.

The one of the typical irradiation experiments was performed to adjust nuclear physical constants for actinides. Two pins including eight quartz capsules containing pure actinides (outer diameter: 4.5 mm, length: 20-21 mm, thickness: 0.7-0.8 mm), shown in Table 28, were prepared in RIARR. These pins were installed into an experimental assembly having 19 pins, as shown in Figure 38. The irradiation was started in December 1998. The irradiation of one pin was finished in October 2001.

4.3.2.4 JOYO and MONJU

- JOYO

The experimental fast reactor JOYO of the Japan Atomic Energy Agency (JAEA) is the first sodium-cooled fast reactor (SFR) in Japan. JOYO attained its initial criticality as a breeder core (MK-I core) in 1977. During the MK-I operation, which consists of two 50 MWt and six 75 MWt duty cycles, the basic characteristics of plutonium (Pu) and uranium (U) mixed oxide (MOX) fuel core and sodium cooling system were investigated and the breeding performance was verified. In 1983, the reactor increased its thermal output up to 100 MWt in order to start the irradiation tests of fuels and materials to be used mainly for other SFRs. Thirty-five duty cycle operations and many irradiation tests were successfully carried out using the MK-II core by 2000. The core was then modified to the MK-III core in 2003 [62, 63]. In order to obtain higher fast neutron flux, the core was modified from one region core to two region core with different Pu fissile contents. Accordingly, the reactor power increased up to 140 MWt together with a renewal of intermediate heat exchangers (IHXs) and dump heat exchangers (DHXs). Table 29 shows the main parameters of the MK-III core. The rated power operation of the MK-III core started in 2004. The MK-III core has been used for the irradiation tests of fuels and materials for future FBRs including prototype FBR MONJU and other R&D fields like transmutation technology as well. This powerful neutron irradiation flux has an advantage, especially for high burn-up fuel irradiation and material irradiation with high neutron dose.

As an example, Figure 39 shows the configuration of the MK-III 3rd operational cycle core in 2006. The fuel region is divided into two radial enrichment zones to flatten the neutron flux distribution. The MK-III driver fuel is a MOX fuel with about 18 wt% enriched uranium. The fissile Pu content ($^{239}Pu+^{241}Pu$)/(U+Pu)) is about 16 wt% in the inner core fuel and about 21 wt% in the outer core fuel. The fuel region is surrounded by a radial stainless steel reflector region, which is about 25 cm thick. Shielding subassemblies including B4C are loaded in the outer two rows of the reactor grid.

Figure 39: Configuration of JOYO MK-III 3rd operational cycle core

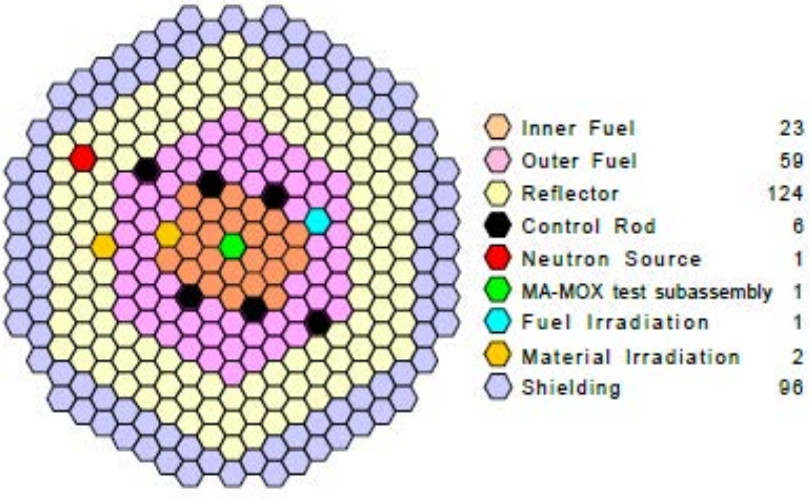


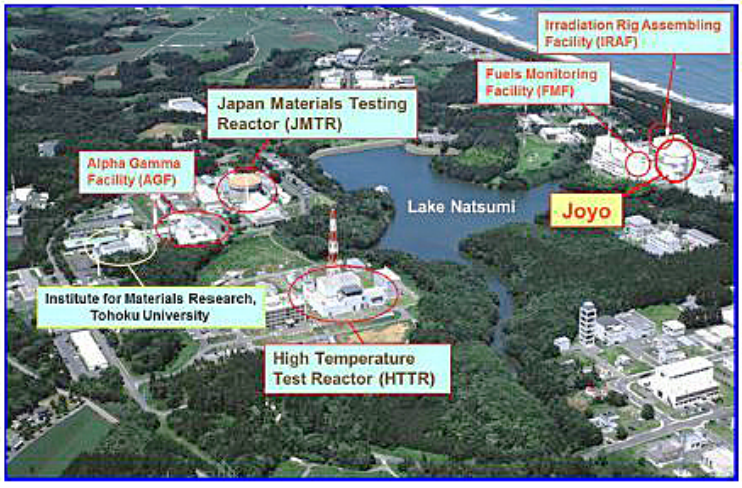
Table 29: Main JOYO core and plant parameters

Reactor thermal output	(MWt)	140
Max. No. of irradiation test S/A		21
Core diameter	(cm)	80
Core height	(cm)	50
^{235}U enrichment	(wt%)	18
Pu content	(wt%)	≤30
Pu fissile content (inner/outer core)	(wt%)	～16/21
Neutron flux total	($n/cm^2 \cdot s$)	5.7×10^{15}
Fast (E>0.1MeV)	($n/cm^2 \cdot s$)	4.0×10^{15}
Primary coolant temp. (inlet/outlet)	(C)	350/500
Operation period	(days/cycle)	60
Reflector/Shielding		SUS/B_4C
Max. excess reactivity (at 100C)	(%Δk/kk')	4.5
Control rod worth	(%Δk/kk')	≥7.6

JOYO provides the maximum fast neutron flux of 4.0×10^{15} n/cm^{2}· s that is the highest fast neutron flux in the irradiation test facilities in the world. The irradiation rigs are grouped under offline irradiation rigs which need off-line monitors to measure irradiation data, and the online irradiation rigs which allow online data measurement and/or temperature control. JOYO can irradiate various fuels and materials. Additionally, JOYO can conduct open core irradiation tests with possibility of a fuel failure based on a license for power-to-melt (PTM) and run-to-cladding-breach (RTCB) test.

There are some facilities necessary for post-irradiation examinations (PIEs) at Oarai Research and Development Centre, as illustrated in Figure 40. The irradiated test subassemblies have been transferred to the Fuel Monitoring Facility (FMF) adjacent to JOYO. The FMF has the large stainless steel-lined, nitrogen-gas-tight hot cells with the remote maintenance systems for conducting PIEs. Additionally, the FMF can conduct non-destructive PIEs using the X-ray computer tomography (CT) device. The test subassemblies are disassembled in FMF and the capsules are retrieved which contain several set of multiple activation foils and helium accumulation fluence monitors (HAFMs) to be used for the neutron dosimetry [64]. The activation foils are finally transferred to the Irradiation Rig Assembling Facility (IRAF) adjacent to JOYO where the gamma-ray measurement system is installed to analyse the reaction rate of the irradiated foils. The retrieved fuel samples are transferred to the Alpha Gamma Facility (AGF) to conduct the destructive PIE to determine the fuel burn-up by means of ^{148}Nd method and to take the metallography of cut cross-section of the samples as an example. The AGF was originally established as the PIE Facility of the irradiated fuels, and later a small-scale fuel fabrication unit was equipped in the hot cell of the AGF to develop a remote fuel fabrication technology of MA-MOX fuel.

Figure 40: Facilities in Oarai R&D Centre



JOYO has a large potential not only for the SFR development but also for a variety of users ranging from fusion, other concepts of next generation reactors (Gen-IV) and ADS as well. JOYO is the only test reactor in Japan which has licenses to conduct the open core PTM or RTCB test and to irradiate fuel pins of which physical or material property is not well known. The irradiation temperature is now able to extend for both lower and higher values from the normal operation temperatures of SFR. Furthermore, higher neutron flux level in the world as much as 4.0×10^{15} n/cm^{2}· s provides the accelerated irradiation to meet with the request of many users. Currently, JOYO has not been operating due to the in reactor-vessel trouble which occurred in 2007 [65]. However, the resumption work has been implemented, and it is planned to be completed in 2014. JOYO is expected to be used as a powerful irradiation facility after the restart.

- MONJU

MONJU is a core R&D facility for the FBR development in Japan. The operating experience of MONJU will be valuable evidence of plant reliability of FBRs and contributes to the establishment of real operating techniques including sodium handling technology. The experimental data obtained from the power operation of MONJU will be used to validate the design of MONJU itself and to verify many analytical codes for future plant design. The construction of MONJU started in 1985. The first criticality was attained in April 1994, and MONJU was temporarily connected to the grid on 29 August 1995. On 8 December 1995, a sodium leak occurred from the secondary circuit, and then MONJU suspended its operation and restarted in May 2010 after 14 years and five months suspension.

MONJU is a sodium-cooled fast breeder reactor with 280 MW electricity and 714 MW thermal output. It has a loop-type reactor system with three primary and secondary loops. The core consists of two fuel regions (Inner and Outer) surrounded by blanket fuel and shielding. It uses plutonium-uranium mixed oxide fuel. The fissile plutonium content in the driver fuel is 16% in the inner core and 21% in the outer core. The core fuel column length is 930 mm and the core equivalent diameter is around 1 800 mm. The driver fuel pin diameter is 6.5 mm and the fuel pellet is solid, with 85% theoretical density.

The largest difference between the restart core and the previous core, a core before the interruption, is in the contents of ^{241}Pu and ^{241}Am. The content of ^{241}Pu has halved and that of ^{241}Am has more than doubled due to the ^{241}Pu decay during the interruption. The restart core contained about 1.5% of ^{241}Am in the whole core average [66]. It is expected that the reactor physics data taken in the restart core together with those in the previous core would be useful to verify the nuclear data of the two nuclides. In particular, verification or improvement of ^{241}Am capture cross-section data is of strong importance since the data still have large uncertainty even in the recent nuclear data. An impact of the reactor physics data obtained in the restart core is investigated by the cross-section adjustment technique with JENDL-3.3 and JENDL-4.0. Criticality data obtained before and after the interruption are applied. It is confirmed that MONJU reactor physics data, when the two data are used together, effectively adjust ^{241}Am capture cross-sections. From these results, the high prediction accuracy for the criticality was demonstrated and the verification data for nuclear data was acquired.

Further developmental effort will be made, which will include fuel property study and fuel pin bundle irradiation testing, as well as fuel pin irradiation tests. International collaborations are indispensible to promote fast reactor fuel development. Generation IV International Forum (GIF) is a relevant framework of such collaborations. One such collaboration is the Global Actinide Cycle International Demonstration (GACID) which includes CEA France, USDOE and JAEA Japan, under the GIF sodium-cooled fast reactor (SFR) system arrangement [67]. The GACID project aims to demonstrate MA transmutation capability and MA-bearing fuel integrity in a fast reactor core, using JOYO and MONJU. The project consists of three steps as listed below and a series of irradiation tests in JOYO and MONJU:

Step 1: Precedent Limited MA-bearing Fuel Preparatory Irradiation Test. This test assumes ^{237}Np and ^{241}Am only as for the MA. Moreover, only a single pin-scale irradiation test at MONJU is planned. Therefore, this test is expected to be implemented at an earlier stage of the project.

Step 2: Pin-scale Cm-bearing Fuel Irradiation Test. A full range of MA composition is assumed for this test. Not only Np and Am but also Cm will be contained in the test fuel, although the test will be conducted on a pin scale. A precedent irradiation test at JOYO is being planned for MONJU irradiation licensing.

Step 3: Bundle-scale MA-bearing Fuel Irradiation Demonstration. After completing the above two steps of the precedent irradiation tests, the final goal, bundle-scale full-range

MA-bearing fuel irradiation demonstration, will be performed at MONJU. This technical demonstration will be undertaken in a reasonable time frame, and the whole project is to be conducted over a period of 20 years.

MONJU is expected to play an important role, as is JOYO, to demonstrate the technical feasibility of TRU homogeneous recycling fuels, the future programme of MONJU is under discussion.

4.3.2.5 ATR and NRAD

- ATR

The ATR is located at the ATR Complex on the INL site and has been operating continuously since 1967. The primary mission of this versatile facility was initially to serve the US Navy in the development and refinement of nuclear propulsion systems, however, in recent years the ATR has been used for a wider variety of government- and privately-sponsored research.

The designation of the ATR as a National Scientific User Facility (NSUF) provides nuclear energy researchers access to world class facilities to support the advancement of nuclear science and technology. The ATR NSUF accomplishes this mission by offering state-of-the-art experimental irradiation testing and PIE facilities and technical assistance in design and safety analysis of reactor experiments. The ATR has large test volumes in high-flux areas. Designed to permit simulation of long neutron radiation exposures in a short period of time, the maximum thermal power rating is 250 MWth with a maximum un-perturbed thermal neutron flux of 1.0x1015 n/cm^2–s, as shown in Table 30. Since most contemporary experimental objectives generally do not require the limits of its operational capability, the ATR typically operates at much lower power levels. Occasionally, some lobes of the reactor are operated at higher powers that generate higher neutron flux.

The ATR is cooled by pressurised (2.5 MPa [360 psig]) water that enters the reactor vessel bottom at an average temperature of 52°C (125°F), flows up outside cylindrical tanks that support and contain the core, passes through concentric thermal shields into the open upper part of the vessel, then flows down through the core to a flow distribution tank below the core. When the reactor is operating at full power, the primary coolant exits the vessel at a temperature of 71°C (160°F). The unique design of the ATR (Figure 41) control devices permits large power variations among its nine flux traps using a combination of control cylinders (drums) and neck shim rods. The beryllium control cylinders contain hafnium plates that can be rotated towards and away from the core, and hafnium shim rods, which withdraw vertically, can be individually inserted or withdrawn for minor power adjustments. Within bounds, the power level in each corner lobe of the reactor can be controlled independently to allow for different power and flux levels in the four corner lobes during the same operating cycle.

Neutron flux in the ATR varies from position to position and along the vertical length of the test position. It also varies with the power level in the lobe(s) closest to the irradiation position. Thermal and fast flux intensity values listed in Table 31 are at the core midplane for a reactor power of 110 MWth and assume a uniform reactor power of 22 MWth in each lobe.

Table 30: ATR general characteristics

Reactor:	
Thermal power	250 MW_{th} [a]
Power density	1.0 MW/L
Maximum thermal neutron flux	$1.0x10^{15}$ n/cm^2-sec [b]
Maximum fast flux	$5.0x10^{14}$ n/cm^2-sec [b]
Number of flux traps	9
Number of experiment positions	68 [c]
Core:	
Number of fuel assemblies	40
Active length of assemblies	4 feet
Number of fuel plates per assembly	19
Uranium-235 content of an assembly	1,075 g
Total core load	43 kg [d]
Coolant:	
Design pressure	2.7 Mpa (390 psig)
Design temperature	115°C (240°F)
Reactor Coolant:	
Light water maximum coolant flow ra	3.09 m^3/s (49,000 gpm)
Coolant temperature (operating)	<52°C (125°F) inlet, 71°C (160°F) outlet

a. Maximum design power. ATR is seldom operated above 110 MWth
b. Parameters are based on the full 250 MWth power level and will be proportionally reduced for lower reactor power levels.
c. Only 66 of these are available for irradiations.
d. Total U-235 always less due to burn-up.

Figure 41: ATR core cross-section

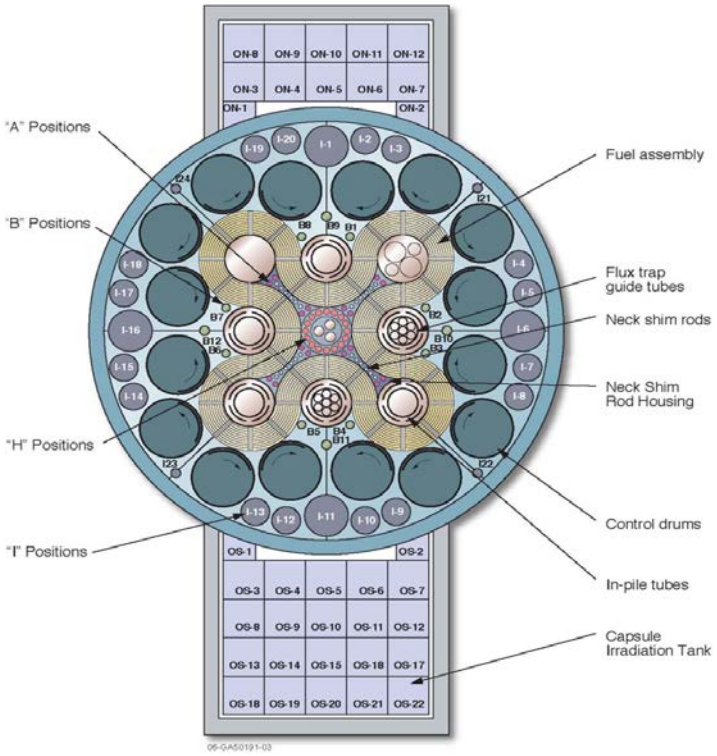


Table 31: Approximate peak flux values for ATR capsule positions at 110 MWth (22 MWth in each lobe)

	Positions	Diameter (in.)[a]	Thermal Flux (n/cm^2-s)[b]	Fast Flux (E>1 MeV) (n/cm^2-s)	Typical Gamma Heating W/g (SS)[c]
Northwest and Northeast Flux Traps		5.250	$4.4x10^{14}$	$2.2x10^{14}$	
Other Flux Traps		3.000[d]	$4.4x10^{14}$	$9.7x10^{13}$	
A-Positions					
	(A-1 - A-8)	1.590	$1.9x10^{14}$	$1.7x10^{14}$	8.8
	(A-9 - A-12)	0.659	$2.0x10^{14}$	$2.3x10^{14}$	8.8
*	**(A-13 - A-16)**	**0.500**	**$2.0x10^{14}$**	**$2.3x10^{14}$**	8.8
B-Positions					
*	**(B-1 - B-8)[f]**	**0.875**	**$2.5x10^{14}$**	**$8.1x10^{13}$**	6.4
*	**(B-9 - B-12)**	**1.500**	**$1.1x10^{14}$**	**$1.6x10^{13}$**	5.5
H-Positions					
	(H-1 - H-16)	0.625	$1.9x10^{14}$	$1.7x10^{14}$	8.4
I-Positions					
*	**Large (4)**	**5.000**	**$1.7x10^{13}$**	**$1.3x10^{12}$**	0.66
*	**Medium (16)**	**3.500**	**$3.4x10^{13}$**	**$1.3x10^{12}$**	
*	**Small (4)**	**1.500**	**$8.4x10^{13}$**	**$3.2x10^{12}$**	
Outer Tank Position					
	ON-4	Var[e]	$4.3x10^{12}$	$1.2x10^{11}$	0.15
	ON-5	Var[e]	$3.8x10^{12}$	$1.1x10^{11}$	0.18
	ON-9	Var[e]	$1.7x10^{12}$	$3.9x10^{10}$	0.07
	OS-5	Var[e]	$3.5x10^{12}$	$1.0x10^{11}$	0.14
	OS-7	Var[e]	$3.2x10^{12}$	$1.1x10^{11}$	0.11
	OS-10	Var[e]	$1.3x10^{12}$	$3.4x10^{10}$	0.05
	OS-15	Var[e]	$5.5x10^{11}$	$1.2x10^{10}$	0.20
	OS-20	Var[e]	$2.5x10^{11}$	$3.5x10^{9}$	0.01

a. Position diameter. Capsule diameter must be smaller

b. Average speed 2,200 m/s.

c. Depends on configuration

d. Current east, center, and south flux trap configurations contain seven guide tubes with inside diameters of 0.694 in.

e. Variable; can be either 0.875, 1.312, or 3.000 in.

f. B-7 is the location of the Hydraulic Shuttle Irraidation System

* Positions available for experiment irradiation in FY-2009

- NRAD

The neutron radiography (NRAD) reactor is a TRIGA® (Training, Research, Isotopes, General Atomics) Mark II tank-type research reactor located in the basement, below the main hot cell, of the Hot Fuel Examination Facility (HFEF) at the Idaho National Laboratory (INL). It is equipped with two beam tubes with separate radiography stations for the performance of neutron radiography irradiation on small test components.

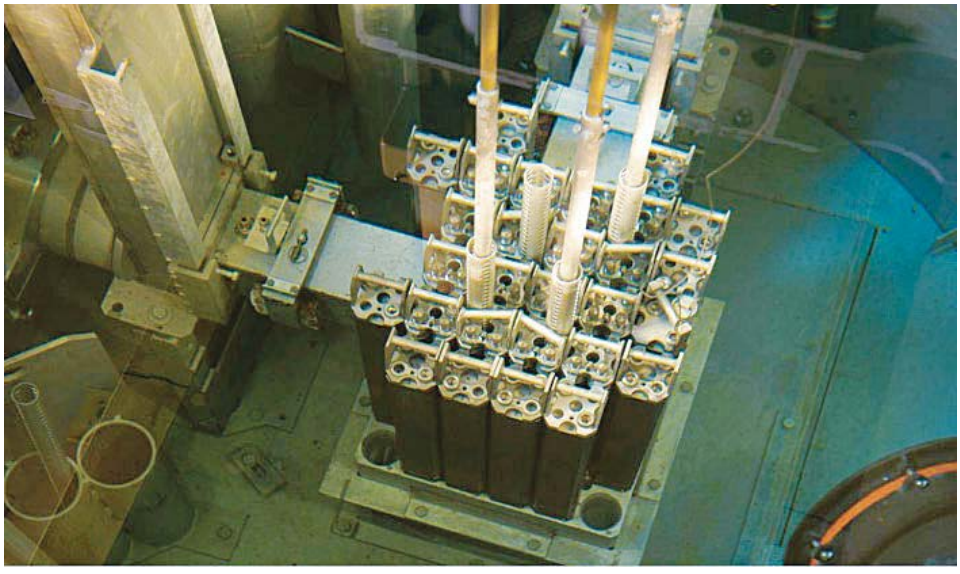
The NRAD reactor is currently under the direction of the Battelle Energy Alliance (BEA) and is operated and maintained by the INL and Hot Cell Services Division. It is primarily used for neutron radiography analysis of both irradiated and un-irradiated fuels and materials. Typical applications for examining the internal features of fuel elements and assemblies include fuel pellet separations, fuel central-void formation, pellet cracking, evidence of fuel melting, and material integrity under normal and extreme conditions.

The NRAD core is designed for steady-state operation with or without in-core and/or in-tank experiments. The combined reactivity worth of all removable experiments within the reactor tank is limited to less than \$0.50.

The NRAD reactor is a TRIGA-conversion-type reactor originally located at the Puerto Rico Nuclear Centre (PRNC). It was converted to a TRIGA-FLIP-(Fuel Life Improvement Programme)-fuelled system (70% ^{235}U) in 1971. The 2-MW research reactor was closed in 1976 and then a portion of the TRIGA reactor fuel elements and other components (with a single radiography beam line) were moved in 1977 by the US Department of Energy (DOE) to Argonne National Laboratory (West) in Idaho Falls, Idaho. The NRAD reactor was first brought to critical in October 1977, and then became operational in 1978. A second beam line was added in 1982.

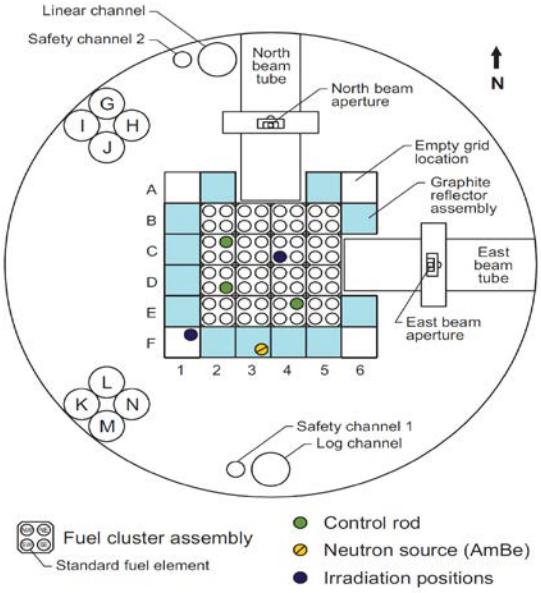
The NRAD reactor (Figure 42) is a 250 kW TRIGA LEU conversion reactor that is a water-moderated, heterogeneous, solid-fuel, tank-type research reactor. The reactor is composed of fuel in three- and four-element clusters that can be arranged in a variety of lattice patterns, depending on reactivity requirements. The grid plate consists of 36 holes, on a 6-by-6 rectangular pattern that mate with the end fittings of the fuel cluster assemblies.

Figure 42: In-tank view of the NRAD reactor core



The NRAD LEU core configuration contains 60 fuel elements, two water-followed shim control rods, and one water-followed regulating rod (Figure 43). A water hole is provided as an experimental irradiation position. The NRAD reactor uses graphite neutron reflector assemblies located along the periphery grid plate locations. The number and position of fuel-element and reflector assemblies can be varied to adjust core reactivity.

Figure 43: NRAD LEU core configuration



Possible Future Minor Actinide Experimental Campaigns at ATR and NRAD

The MANTRA (Measurements of Actinide Neutron Transmutation Rates with Accelerator mass spectrometry) is the first reactor physics integral experiment performed in the US in more than 20 years. The experiment is performed at the ATR Facility with the objective of providing valuable experimental data on neutron cross-sections for advanced fuel cycles. Irradiation of samples of several actinides and a few fission products using three separate neutron filters to simulate different spectra is underway.

Even though the MANTRA experimental programme is expected to be a success, there is already a need for a second phase (a MANTRA-2) of such a type of experiment. There are several reasons that can justify this request. First, there are several actinide samples that, for different reasons, have not been irradiated among others: ^{238}Pu, ^{241}Pu, and especially ^{244}Cm (irradiated only with thin filters). Moreover, at the time of the second campaign, efficient mass separators should be available at INL. This should allow purifying also samples of isotopes already irradiated in MANTRA and avoiding one of the main concern, related to contamination from other isotopes, in the post-irradiation analysis.

Finally, due to the limited room available in most cases, only one sample per isotope (and in a couple of cases two) will be irradiated in MANTRA. In the PROFIL French experiments, there were three and even six samples (PROFIL-2) of the same isotopes that were irradiated, the reason being that in certain cases in the post-irradiation analysis, due to bad manipulation, some samples were contaminated. We can expect a low rate of failure for MANTRA, but, in case of an emergency, the MANTRA-2 campaign should provide an opportunity for repeating the compromised irradiation of some isotopes.

In complement to the MANTRA campaign, a separate experimental programme can be foreseen to be performed at the NRAD Facility. The INL NRAD is a TRIGA reactor that has enough room to allow the presence of thick neutron filters (including ^{238}U blocks) that will really simulate a large range of spectra going from thermal, epithermal, soft fast, and hard fast. Measurement of fission rate spectral indices using fission micro-chamber can allow the assessment of the knowledge on a vast range of actinides (major and minors). Moreover, reactivity sample oscillations measurements with apparatus that are been implemented at the ISU (Idaho State University) reactors (including open and closed loop) and easily installed at NRAD can be foreseen. Measurements of actinides samples in different spectra will be invaluable for the validation of cross-sections useful for advanced fuel cycles.

4.3.2.6 JHR

The Jules Horowitz Reactor (JHR) is a 100 MW material testing and experimental research reactor currently under construction at CEA Cadarache. It will replace the OSIRIS reactor. The reactor will also produce medical radioisotopes. The first divergence is expected in 2016. The JHR development is managed by a consortium associating public research organisations, utilities, and industry. It will be operated as an international user facility. The JHR will provide flexible irradiation capabilities to study materials and fuel components in well-controlled highly-instrumented flux, temperature and pressure conditions. Several sample-holding devices are under development for irradiation in the core or in the beryllium reflector, including devices (CALIPSO), in which the local environment of an SFR is reproduced (www-cadarache.cea.fr/rjh).

4.3.2.7 VENUS-F zero power facility for critical and subcritical fast experiments

- 4.3.2.7.1 Description of the VENUS-F Facility with GENEPI accelerator

Critical assemblies in the zero-power thermal neutron water-moderated facility VENUS at SCK•CEN (Mol, Belgium) has often served for LWR benchmarking in the past (e.g. [68]). It has been recently decided at SCK•CEN to modify the facility in order to comply with internationally identified needs for new integral experiments in zero-power facilities. At a European level, the importance of developing lead-based fast systems and R&D activities in support of accelerator-driven systems has been highlighted. This requires integral experiments in a representative zero-power facility. Therefore, in the period 2007-2010, the thermal neutron water-moderated facility VENUS was modified into a fast neutron facility VENUS-F with solid lead core components (Figure 44).

The assembly top section of the VENUS-F reactor contains now 12x12 assembly grid (i.e. the reactor core), in which different types of assemblies and rods can be loaded: fuel assemblies, lead assemblies, safety rods, control rods, experimental assemblies (Figures 45 and 46).

Figure 44: Specifications of VENUS-F Facility

Experimental equipment	Wide list of calibrated fission chambers also with minor actinides, large number of activation foils, automatic scanner with several Ge detectors, GENEPI-3C 250kV D-T accelerator giving 14 MeV neutrons
Possibilities for GIV research	Fast reactors designs (LFR, SFR and ADS) mock-up and benchmarking
Max flux (fast)	10^{10} n cm^2 s^{-1}
Vessel height	203 cm
Core volume	120 dm^3
Core height	120 cm
Reflector thickness	40 cm
Reflector material	Pb, SS, Bi, UO_2 depleted
Fuel type	U, MOX
Fuel enrichment	7-30
Fuel currently available	U 30% rodlets 12.7x200 mm
Scheme	

Figure 45: Radial section through the VENUS-F reactor

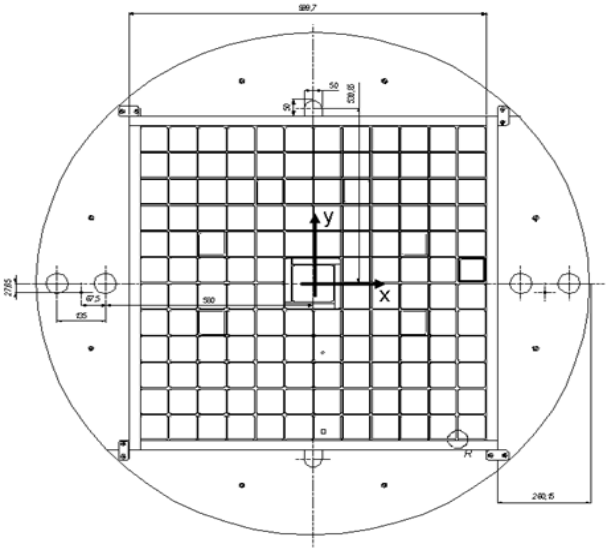
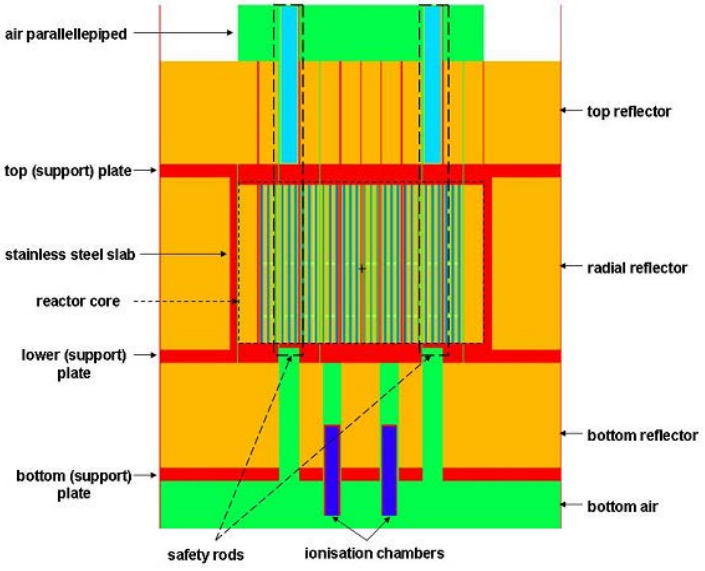


Figure 46: Axial section through the VENUS-F reactor



The First VENUS-F core consists of 97 fuel assemblies (FA) that are arranged in a cylindrical geometry (~80 cm in diameter, 60 cm in height), and are composed of a 5x5 pattern mixture of metallic uranium fuel and solid lead rodlets to simulate the coolant of a fast reactor system. Lead plates were added around the pattern to decrease the fuel-to-coolant ratio. The outer section of a FA is 80 mm x 80 mm and the pattern chosen is shown in Figure 47. Figure 48 shows the axial cut of FA. The fuel is 30 % ^{235}U enriched metallic uranium (provided by CEA, France). The core is surrounded by two 40 cm axial and radial lead reflectors (Figures 49 and 50).

Figure 47: Radial cut through active fuel part of the VENUS-F fuel assembly

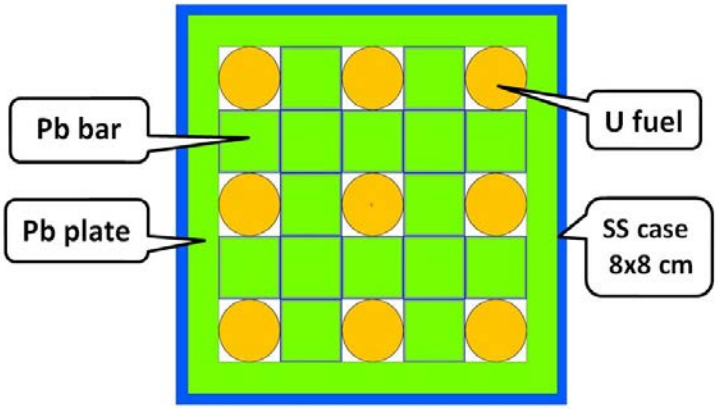


Figure 48: Axial cut through the standard VENUS-F fuel assembly

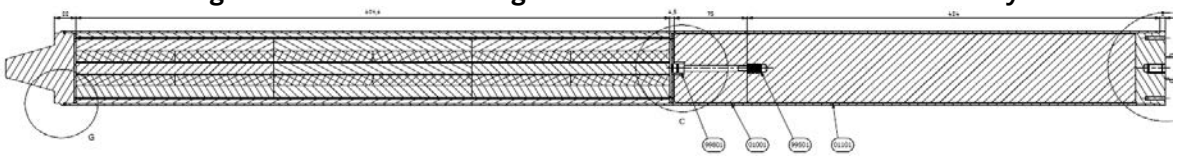
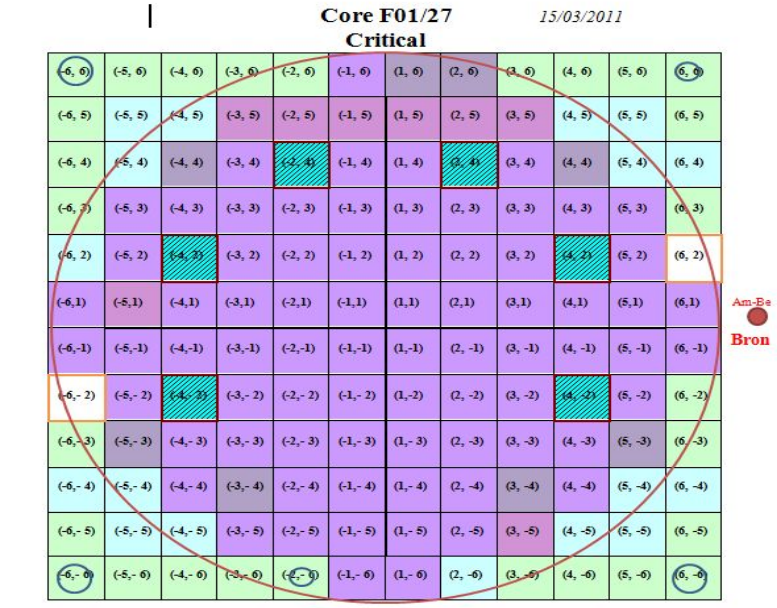
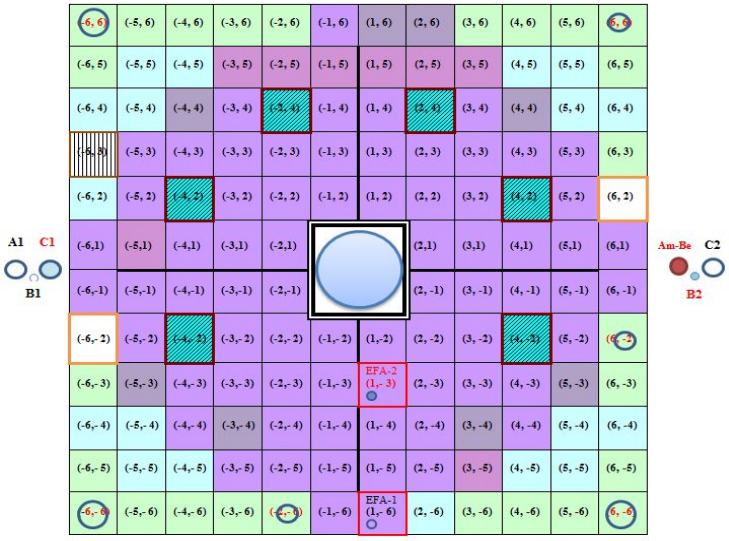


Figure 49: Main critical configuration (core F01/27)



		Amount	SCR,mm	PEAR	POAR
	- lead assemblies, PbA	45			
	- FA	91			
	- standard control rods "SCR"	2	395.4		
	- PEAR rod drop	0		NO	
	- safety rods, SR	6			

Figure 50: Subcritical core configuration (F01/28/02) with vertical accelerator line in the centre



		Amount	SCR,mm	PEAR
	- lead assemblies, PbA	44		
	- FA	87		
	- standard control rods "SCR"	2	479.3	
	- PEAR rod drop	1		IN
	- safety rods, SR	6		

The first project at VENUS-F was GUINEVERE (Generator of Un-interrupted Intense NEutrons at the lead VEnus REactor), an EC FP-6 Project [69] related to ADS research mainly focusing on the validation of a methodology for on-line reactivity monitoring of a

subcritical system. For this purpose, the so-called GENEPI-3C 250 kV deuteron accelerator was designed and built by a CNRS/IN2P3 in this project to couple to the VENUS-F lead core [70]. This accelerator can work both in pulsed mode and continuous mode and is ended by a TiT target providing 14 MeV neutrons via the T(d,n) reaction. The new GENEPI-3C machine cumulates specifications of the first GENEPI accelerator, i.e. pulsed mode operation with very sharp and intense beam pulses (1 µs, 50 mA peak current), with new continuous mode specifications summarised in Figure 51.

In this new DC mode, it will be also possible to operate beam interruptions (so-called "beam trip") with a programmable duration and a low repetition rate for the needs of the foreseen experiments. The accelerator itself consists of an ion source sited in the 250 kV high voltage head, followed by a horizontal beam transport line section of 3 m at the exit of the accelerator tube. Beam transport is ensured with electrostatic quadrupoles. A first group of 4 quadrupoles transports and focuses the beam at the 90° dipole magnet, which deflects the beam downwards. A quadrupole doublet is located at the exit of the magnet chamber and two quadrupole triplets located on the vertical beam line above the reactor core focus the beam onto the target. The target is located at the end of a short optic free thimble inserted into the reactor core (Figures 50, 52).

Figure 51: Specifications of the GENEPI-3C accelerator installed at the VENUS-F site

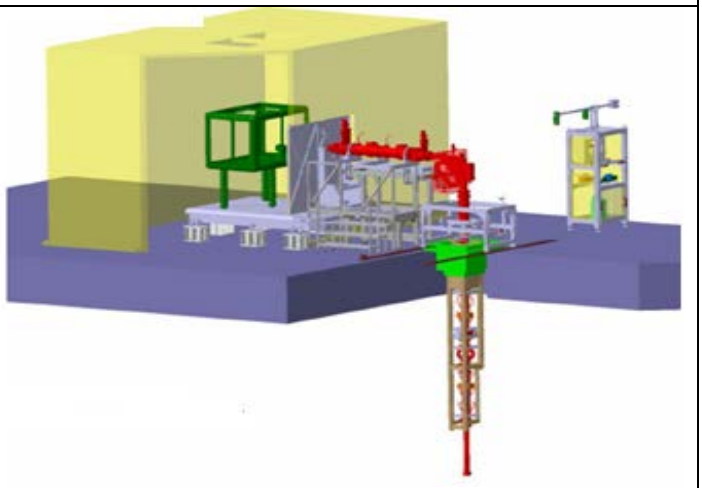
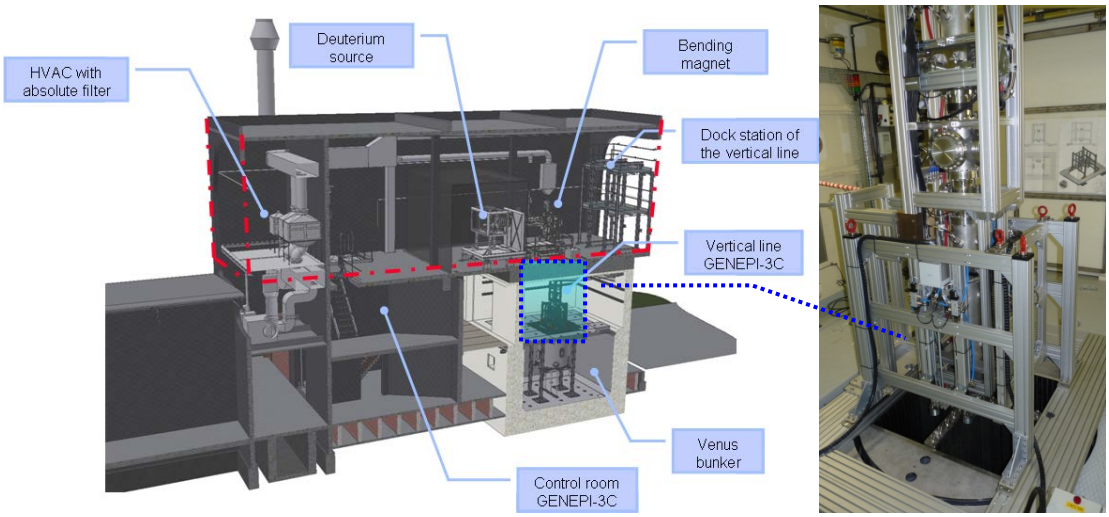
Mean current	160 µA to 1 mA
High Voltage	250 kV
Beam trip rate	0.1 to 100 Hz
Beam trip duration	~ 20 µs to 10 ms
Transition (ON/OFF)	~ 1 µs
Beam spot size	20 to 40 mm in diameter
Maximum neutron production	~5×1010 n/s
Pulse stability	~1%
General layout of the GENEPI-3C machine with its supporting structures, Faraday cage and magnet cooling unit.	

Figure 52: Cross-section of the VENUS site and picture of the insertion of the accelerator vertical beam line in the VENUS-F core



- 4.3.2.7.2 Equipment and measurement techniques used at critical and subcritical VENUS-F cores

4.3.2.7.2.1 Equipment

4.3.2.7.2.1.1 Fission chambers

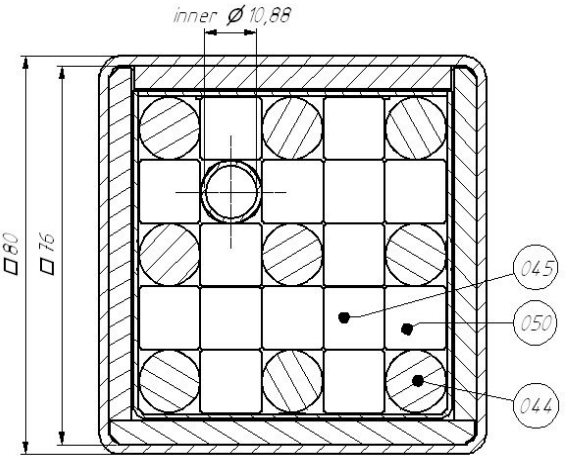
To measure fission rate distributions (traverses), spectrum indexes and monitoring of the flux in critical and subcritical VENUS-F cores, the fission chambers with different sizes, deposit mass and isotopes are used (Table 32).

Table 32: Parameters of fission chambers available at VENUS-F

Isotopes in deposits	^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{242}Pu, ^{237}Np, ^{234}U, ^{232}Th
Mass of deposit	From 10 µg to 1 g
Outer diameter (mm)	1.5, 4, 8, 25, 50

The detailed description of all available fission chambers as well as the possible positions of their placement in the reflector and the core of VENUS-F reactor are presented in [71]. Figure 53 shows the experimental fuel assembly for measurements of spectrum indexes and axial traverses by 1.5, 4 and 8 mm FC. In this experimental FA, the Pb rodlet is replaced on the total height by stainless steel guiding tube. Other details are the same as in the standard FAs.

Figure 53: Experimental fuel assembly with SS guiding tube



4.3.2.7.2.1.2 Foils

A huge amount of foils from different materials and with different geometry parameters are in the VENUS storage. To measure spectrum indexes and traverses the foils are easy placed between fuel and lead rodlets in FAs. A set of the threshold foils can be used for the estimation of the neutron spectrum.

4.3.2.7.2.2 Measurements methods

To determine the power profile of the rector, the ^{238}U, ^{235}U and ^{239}Pu fission rates distributions are measured by 8 mm fission chambers and foils.

The 4 mm, 1.5 mm FCs and foils are used for the determination of the standard spectrum indexes as F238/F235, F239/F235 and C238/F235.

The FCs with minor actinides (MA) are used for the measurements of MA fission ratios to the F235.

β_{eff} and Λ are measured by Rossi-α or other noise method, having the signals from BF_3 ionization chambers.

FCs with big deposit masses (10mg-1g) are used for the monitoring of the neutron flux in subcritical states with external sources (Am-Be or GENEPI accelerator).

To determine the subcritical level referring to the well-known MSM method, the special rod drop system is available at VENUS-F. The rod drop system has the PEAR (PElletised Absorber Rod) rod with 150 pcm antireactivity.

To control and measure the subcritical level in the coupling system VENUS-GENEPI, the following methods, using the counts from several FCs, are applied:

- PNS area method;
- current to flux;
- source jerk;
- fitting of the alpha-decay constant : k-prompt method.

4.3.2.8 Kyoto University Critical Assembly (KUCA)

The Kyoto University Research Reactor Institute is conducting a research project on the accelerator-driven system (ADS) using the fixed field alternating gradient (FFAG) accelerator [72]. The goal of the research project is to establish a next-generation neutron source by introducing a synergetic system comprising a research reactor and a particle

accelerator, and to demonstrate the basic feasibility of ADS as a new neutron source multiplication system using the Kyoto University Critical Assembly (KUCA) coupled with the FFAG accelerator. Prior to ADS experiments with 150 MeV protons generated from the FFAG accelerator, it was necessary to evaluate the neutronic characteristics of ADS and to establish measurement techniques for several neutronic parameters in ADS. For these purposes, a series of ADS experiments with 14 MeV neutrons generated by the deuterium-tritium (D-T) reactions in a Cockcroft-Walton-type accelerator was carried out at the polyethylene solid-moderated and reflected core (A-core) at KUCA. Several neutronic parameters were measured in these experiments, including neutron multiplication, neutron decay constant, reaction rate distribution, neutron spectrum and subcriticality.

The A-core employed in the ADS experiments was essentially a thermal neutron system composed of ^{235}U fuel and polyethylene-moderator and reflector. Due to safety regulations, in both ADS experiments (14 MeV neutrons and 100 MeV protons), the targets (tritium and tungsten, respectively) were located not at the centre of the core but outside the critical assembly.

The world's first injection [72] of spallation neutrons generated by the high-energy proton beams into the KUCA A-core was successfully accomplished on 4 March 2009. By combining the FFAG accelerator with the A-core, the ADS experiments were carried out under the condition that the spallation neutrons were supplied to a subcritical core through proton injection (100 MeV energy; 30 Hz repetition rate; 10 pA intensity) onto a tungsten target (80 mm diameter and 10 mm thickness).

A series of ADS experiments with 14 MeV neutrons and spollation neutrons by 150 MeV proton were carried out at KUCA. The objective of these experiments was to conduct a feasibility study on ADS from the reactor physics viewpoint, in order to develop an innovative nuclear reactor for a high-performance transmutation system with power generation capability or a new neutron source for scientific research.

Additionally, at KUCA, the thorium-loaded ADS experiments were conducted for the investigation of feasibility of neutron multiplication by thorium fission reactions obtained by high-energy neutrons through the injection of high-energy protons from the FFAG accelerator.

On the basis of the experiments conducted in the past at KUCA relating the minor actinides (MA) of ^{237}Np and ^{241}Am, the consequent ADS experiments with 150 MeV protons are planned to investigate MA characteristics through their reaction rate analyses, and to examine the feasibility of ADS with MA in the high-energy neutron spectrum combined with several fuels (highly-enriched uranium, natural uranium and thorium) and reflectors (polyethylene, aluminium, graphite and beryllium).

4.3.2.9 BFS

- The BFS-1 facility

The BFS-1 Facility was built in the IPPE for full-scale modelling of fast-reactor cores, blankets, in-vessel shielding, and storage. Figures 54 and 55 show the horizontal and vertical cuts of the facility. The core volume can be varied from tens of liters to two cubic meters. The facility started operation in June 1961. Since then, core prototypes of many reactor designs have been studied at this facility and many experiments for validation of neutron data have been performed.

Figure 54: BFS-1 Facility (horizontal cut)

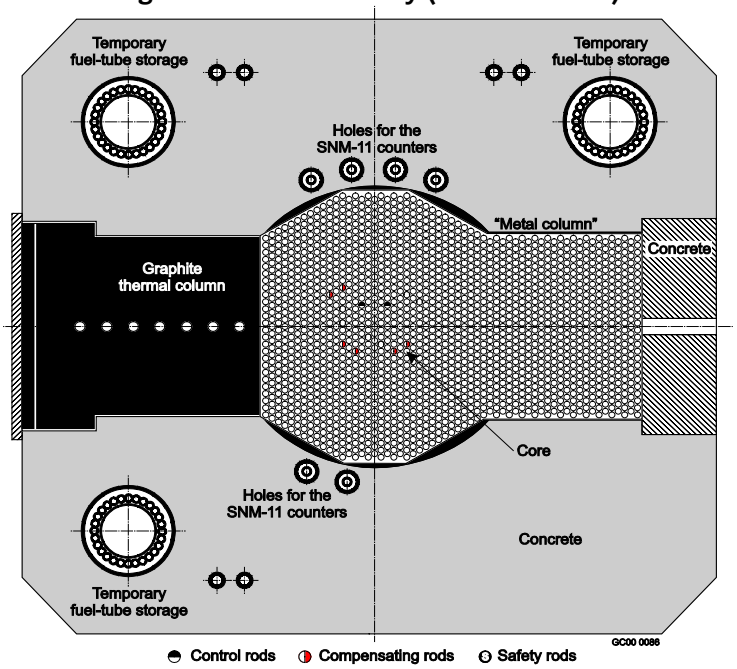
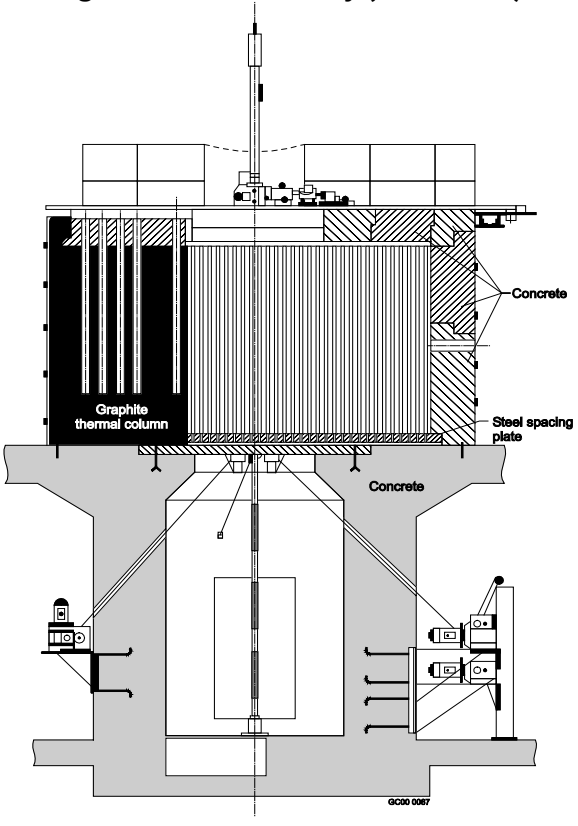
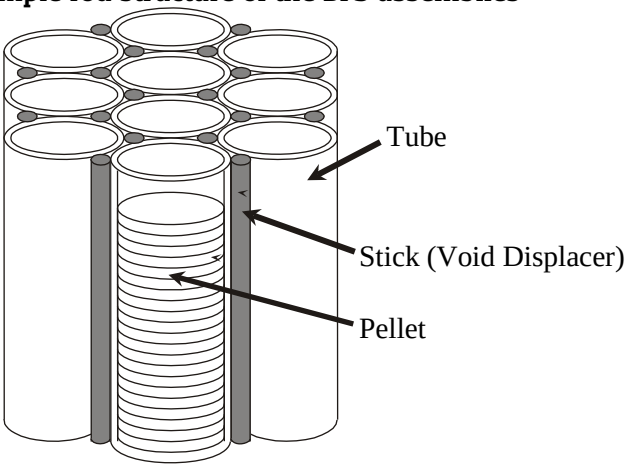


Figure 55: BFS-1 Facility (vertical cut)



The critical assemblies are constructed at the BFS-1 Facility from set of vertical stainless steel or/and aluminum tubes with outer diameter of 5.0 cm and 0.1-cm wall thickness. The hexagonal lattice pitch of the tubes is 5.1 cm. The tubes are filled with disks (pellets) of fissile or structural materials in a repeated cell arrangement. Figure 56 shows the sample geometrical structure of the BFS assembly tubes.

Figure 56: Sample rod structure of the BFS assemblies



These tubes are put inside a cylindrical steel vessel with inner diameter of 2 002 mm, 2 775 mm height, and 22-mm wall thickness. On one side, the cylindrical core vessel is "cut" by an aluminum wall with a thickness of 10 mm. A graphite "thermal column" is behind the wall. The distance from the centre of the vessel to this wall is 778 mm. Cadmium lining (1.00 ± 0.06 mm thick) covers the outer surface of the wall between the aluminum wall and graphite. The graphite thermal column is shown on the left side of Figures 54 and 55.

On the opposite side of the core is a bay for the "metal column", which is an extension of the reflector designed to study neutron penetration through thick layers of materials. The "metal column" is continuous with the central hexagonal grid. The inner width of the bay is 1 327 mm. The distance from the centre of the assembly to the far wall of the bay is 1 836 mm. Cadmium lining (1.00 mm thick) covers the outer surfaces of the 22-mm-thick steel bay walls.

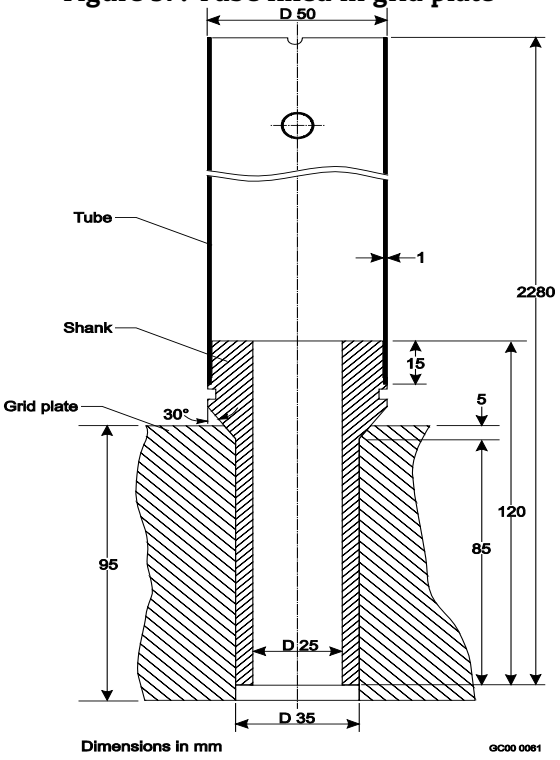
Inside the curved walls of the vessel, steel segments with a cylindrical outer surface (radius 1 000 mm) and flat inner surfaces are fastened. Therefore, the shape of the BFS-1 working volume is an irregular octagonal prism. Cadmium sheets with a thickness of 1.00 mm are placed between the segments and the curved walls of the vessel. Two types of segments are used: two small segments with maximum thickness (arc height) of 29 mm and four big segments with maximum thickness (arc height) of 45 mm. Figure 54 shows the six segments located on a 95-mm-thick steel lattice plate. The height of segments is 2 235 mm.

Large concrete blocks of about 2 g/cm^3 density provide biological shielding around the vessel. The upper layer is made of high-density (approximately 4 g/cm^3) concrete blocks. There is concrete shielding above the thermal column and removable shielding above the "metal column." The concrete cover above the "metal column" was removed during the early stages of the approach to criticality in order to easily move depleted-uranium dioxide tubes from the core to the metal column. Later, it was closed so that the experimentalists could more easily walk around the core. The block masonry forms a cylindrical cavity (206 cm in diameter) for installation of the critical assembly vessel and two rectangular cavities for the graphite and metal columns.

Six SNM-11 type BF_3 impact counters are located near the stainless steel tank containing the critical assembly. The centres of the 15-cm-diameter holes for the counters are about 20 cm from the core vessel. Figure 54 shows also three large cylindrical regions (60 cm diameter) that are used to store fresh fuel tubes. The regions are about 80 cm away from the columns of the assembly.

A 9.5-cm-thick steel support and spacing plate, shown in Figure 55, is located at the bottom of the cylindrical section of the vessel and "metal column". There are 3.5-cm-diameter holes in this grid plate that are drilled in a 5.1 cm hexagonal lattice pitch. Shanks of the tubes forming the assembly are put into these holes (Figure 57). A maximum of 1169 tubes can be installed in the octagonal section of the vessel with 612 additional tubes in the "metal column".

Figure 57: Tube fixed in grid plate



The 14 tubes used as control rods are shown in the central part of Figure 54. The 14 holes in the grid plate that are provided for the control rods have a diameter of 5.3 cm. Control-rod tubes have the same diameter as other tubes of the assembly and similar contents, but they have a longer active length and are connected to drives at their bottom. Control-rod drives are shown in the lower part of Figure 55. The BFS-1 Facility has three types of control-rod drives: two independent drives of control rods, each connected to one tube; three compensating-rod drives, each connected to two tubes; and three safety-rod drives, each connected to two tubes. When control-rod tubes are in their upper positions, fuel and other contents are at the same levels as those in other tubes. The lower, shutdown position of the tubes is usually 80 to 100 cm below the upper level. (End switches installed at the level optimised for each critical assembly determine the lower positions of the control rods.)

An experimental device used for the placement of fission chambers and other experimental detectors are shown in Figure 55 at the top of the facility, above the assembly.

Two types of neutron detectors are used for reactor control, namely KNK-59 ionisation chambers having gamma compensation, placed in peripheral tubes of the lattice, and SNM-11 boron fluoride counters, placed in special holes in the concrete masonry (Figure 54). Five out of the eight KNK-59 chambers are used as reactor control-system detectors (two are used for power-level current channels related to the reactor safety system, two are used for automatic power-control channels, and one for a control-channel and power-doubling-time safety system). A few tube locations in the vicinity of these chambers are filled with polyethylene cylinders of the same diameter. The three remaining KNK-59 chambers are used in the experiments to determine reactivity effects. SNM-11 neutron counters are used for reactor control during the approach to criticality.

- The BFS-2 Facility

The BFS-2 Facility was designed in the IPPE for the full-size simulation of cores and shielding of large fast reactors with a unit power up to 3 000 Mwe (Figure 58). It started operation in October 1969. In August 1989, it was shut down for a reconstruction of the control room and for changing the electronic equipment of the system of monitoring and protection. It restarted operation in March 1990.

Figure 58: BFS-2 Facility

The BFS-2 critical facility is in the same building as the BFS-1.

The walls of the vessel, in which the investigated critical assemblies are constructed, limit a space of an irregular prismatic form with an effective diameter of 5 metres and a height of 3.3 metres.

The vessel walls are vertical screens, each of them consisting of two steel sheets, 14 mm and 6 mm thick, with the cadmium layer between them. The cadmium layer is 1 mm thick and its purpose is an absorption of neutrons which slow down in the concrete shielding and come back into the vessel of the critical facility.

9460 tubes are installed inside of the vessel. These tubes are made of stainless steel (1X18H10T). Their outside diameter is 50 mm and the wall thickness is 1mm. The tails of the tubes are installed in holes of the diagrid plate. The thickness of this steel plate is 100 mm.

The hexagonal lattice pitch of the diagrid is 51 mm.

The space between the tubes can be filled with stainless steel (1X18H10T) sticks of 8 mm diameter and about 3 100 mm long. This space can be also used for inserting different small-size detectors for experimental investigations of reaction rate distributions.

To carry out calibration measurements requiring the thermal spectrum of neutrons, the critical facility is equipped with the graphite column which is installed closely to the tubes in the vessel. The walls of the thermal column are made of double steel sheets (with thickness 7 mm each) with a 1 mm cadmium sheet between them.

For simulating in-vessel storages, shielding and screens, there is a so-called metal column in the vessel part opposite to the graphite column. The tube lattice pitch in the metal column is the same as in other parts of the vessel. There are 1 330 tubes in this column. Some special cassettes can be installed in the metal column for an extraction of a horizontal neutron beam.

For loading (unloading), changing positions, lifting (lowering) or oscillating fuel and absorber rods in the core as well as for installing detectors in different positions at a power level, the facility is equipped with a coordinate manipulator whose load rating is 100 kg. The grip of the manipulator is guided automatically with necessary accuracy on a pre-selected coordinate for any of 9 460 tube positions. Being operated in a measuring regime, the manipulator is controlled by a computer.

At the storages of BFS facilities, there are large amounts of fissile material (metal and dioxide of uranium 36% and 90% enrichments, weapon and reactor grade metal plutonium, about 8 tonnes), about 280 tonnes of fertile materials, 120 tonnes of structural materials (stainless steel, Al, Ni, Nb, Zr, C, B_4C, $Al_2O_{3..}$) and 200 tonnes of coolant materials (sodium, lead, lead-bismuth). All the reactor materials are in the form of pellets with 47 mm diameter and 10-0.1 mm thickness. Some of the pellets are covered by Al or SS. The same pellets are used for the BFS-1 as well as for the BFS-2 assemblies.

The possibility of the BFS critical facilities allows investigating not only the models of reactors with fast neutron spectrum, traditional for these installations, but also using the pellets of polyethylene as slow-down material and the advanced reactor models LWR type, where the neutron spectrum is essential thermal.

At the BFS facilities, the models of the BOR-60, IBR-2, BN-350, BN-600, BN-800, BN-1200, BREST-300, CEFR (Chinese fast Na cooled), KALIMER (Korean fast Na-cooled), SVBR-100, MBIR-100 core designs were investigated in addition to plenty of testing experiments and cross-section libraries (benchmarks).

4.3.2.10 Transmutation Physics Experimental Facility (TEF-P) in J-PARC

The Japan Atomic Energy Agency (JAEA) has been implementing research and development (R&D) on partitioning and transmutation (P&T) technology to reduce the burden of the back-end of the nuclear fuel cycle. The R&D on P&T in JAEA is based on two different concepts; one is the homogeneous recycling of minor actinide (MA) in commercial fast breeder reactors and the other is the dedicated MA transmutation, the so-called “double-strata” strategy, using an accelerator-driven system (ADS).

The Atomic Energy Commission (AEC) of Japan conducted Check and Review on P&T technology by establishing a subcommittee in August 2008. The subcommittee compiled and submitted a final report to AEC in April 2009. As the basic policy for R&D of P&T technologies, the report emphasises the need for the R&D of double-strata concept as part of the R&D for future nuclear power generation systems considering the transitional phase from light-water reactor (LWR) cycle to FBR cycle as well as the equilibrium phase of FBR cycle. Regarding the ADS, the report recognises the significant progress in the design study, accelerator development, LBE technology including a spallation target, etc. It is, however, pointed out that the ADS is still in the fundamental stage as a whole and that further accumulation of basic data is necessary to step up to the next stage to verify

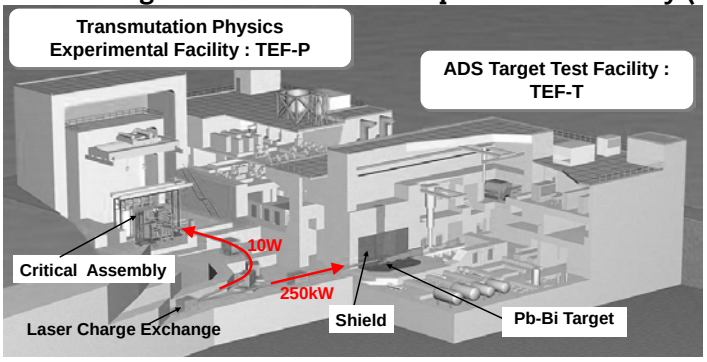
the engineering feasibility. The report highlights the required R&D issues on ADS as follows:

- reliability and economy of the accelerator;
- feasibility of the beam window;
- reactor physics of subcritical core including its control;
- design and safety of LBE cooled core.

The report also emphasises the requirements of the critical experiments using MA for the integral experiments of the transmutation system and recommends that the Transmutation Physics Experimental Facility (TEF-P) planned in the second phase of J-PARC project should be investigated as part of the creation of the environment to perform these experiments.

JAEA is conducting a multi-purpose high-intensity proton accelerator programme, called J-PARC, in collaboration with KEK. To study the basic characteristics of the ADS and to demonstrate its feasibility from the reactor physics and the spallation target engineering viewpoints, JAEA is planning to build the Transmutation Experimental Facility (TEF) [73] in the Tokai site within the framework of the J-PARC Project shown in Figure 59. TEF consists of two buildings: the Transmutation Physics Experimental Facility (TEF-P) and the ADS Target Test Facility (TEF-T). TEF-P is a zero-power critical facility, where a low power proton beam is available to research the reactor physics and the controllability of the ADS. TEF-T is a material irradiation facility, which can accept a maximum 250 kW-400 MeV proton beam into the spallation target of LBE.

Figure 59: Image of Transmutation Experimental Facility (TEF)



Several types of experiments have been performed using existing facilities worldwide in order to investigate the neutronic performance of the ADS. In JAEA, subcritical experiments were carried out at the Fast Critical Assembly (FCA) using a ^{252}Cf and DT neutron source. As there was no experiment aiming at the research and the demonstration of the fast subcritical system combined with a spallation source, TEF-P is designed to cover the fields of R&D for:

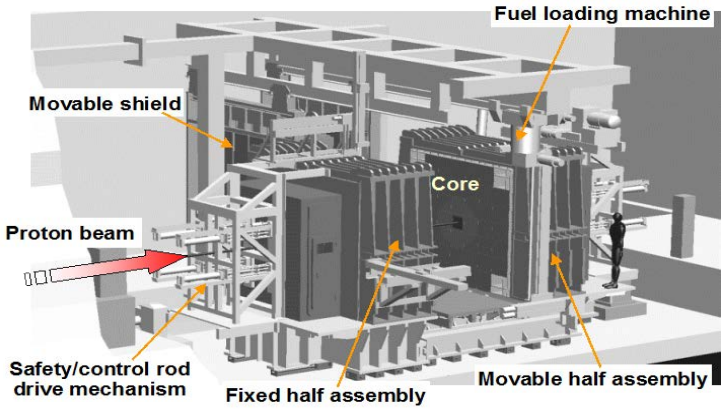
- reactor physics aspects of the subcritical core driven by a spallation source;
- demonstration of the controllability of the subcritical core including a power control by the proton beam power adjustment;
- investigation of the transmutation performance of the subcritical core using certain amount of MA and LLFP.

For these purposes, the high-thermal power is not necessary and a power level of critical experiments such as 100 W is optimal from the viewpoint of the accessibility to the core. Although the thermal feedback effect of the core might be needed, such

experiments can be performed using an electrical heater, which simulates the reactor power without real fission energy and accompanying fission products. The maximum thermal power is temporarily decided as 500 W.

To build a new nuclear facility, fuel fabrication remains a concern, since tonnes of low-enriched uranium or plutonium are required to simulate the ADS (e.g. k_{eff} = 0.95) in the fast neutron system. For this, the plate-type fuel of the FCA can be used in JAEA/Tokai, or to merge FCA into TEF-P. Various simulation materials should be necessary to simulate fast reactors and ADS such as lead and sodium for coolant, tungsten for solid target, ZrH for moderator, B_4C for absorber, and AlN for nitride fuel. TEF-P is designed taking into account FCA, which is a horizontal table-split type critical assembly with a rectangular lattice matrix. Figure 60 shows a conceptual view of the assembly. A proton beam was introduced horizontally from the centre of the fixed half assembly.

Figure 60: Conceptual view of TEF-P



A low current proton beam is extracted by a laser charge exchange technique from a high-intensity beam line of 250 kW (0.63 mA, 400 MeV), most of the beams are introduced into TEF-T. The 250 kW proton beam is pulsed one whose repetition rate is 25 Hz and the maximum pulse width is 0.5 ms. The protons are accelerated as negative ions (H-). The beam is exposed by a YAG-laser, which can strip one of the two electrons to change a small amount (below 10 W) of H- to neutral one (H0). The H- and the H0 are then separated by a bending magnet, where H- is bent into TEF-T and H0 goes straightforward to the magnetic field. The other electron of the H0 is finally stripped by a carbon foil so that the positive protons (H+) are introduced into TEF-P. The time width of the proton pulse for TEF-P can be adjusted by changing the duration of the laser exposure and 1 ns to 0.5 ms pulse is expected to be available. The proton beam intensity can be controlled by a collimator which is installed inside the facility.

Several experimental studies are underway to enable the TEF-P Facility to be operated. Table 33 lists the R&D items that should be carried out at TEF-P [74].

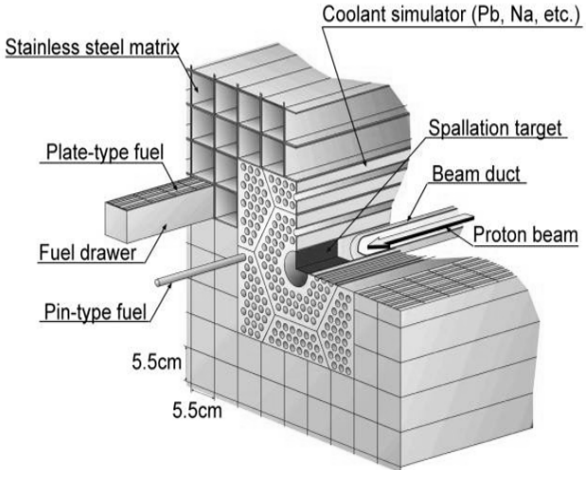
Regarding the neutronics in the subcritical system, power distribution, k_{eff}, effective neutron source strength, and neutron spectrum are measured by changing parametrically the subcriticality and the spallation source position. The material of the target will also be altered with Pb, Pb-Bi, W, etc. The reactivity worth is also measured in the case of the coolant void and the intrusion of the coolant into the beam duct. It is desirable to make the core critical in order to ensure the quality of the experimental data of the subcriticality and the reactivity worth. As for the demonstration of the hybrid system, the feedback control of the reactor power is examined by adjusting the beam intensity. The operating procedures at the beam trip and the re-start are also examined. Regarding the transmutation characteristics of MA and LLFP, fission chambers and activation foils are used to measure the transmutation rates. The cross-section data of MA and LLFP for the high-energy region (up to several hundreds MeV) can be measured

by the Time of Flight (TOF) technique with the proton beam of about 1 ns pulse width. Several types of MA and LLFP samples are also prepared to measure their reactivity worth, which is important for the integral validation of the cross-section data. A fundamental target of the facility is to install a partial mock-up region of MA fuel with air cooling in order to measure the physics parameters of the transmutation system. Figure 61 shows a schematic view of the partial loading of pin-type MA fuel around the spallation target [75, 76]. The central rectangular region will be replaced with a hexagonal subassembly.

Table 33: R&D items to be performed at TEF-P

Purpose	Experimental items
Validation of data and method to predict the neutronics in a fast subcritical system with a spallation source	Measurement of power distribution in subcritical system
	Determination of k_{eff} and effective source strength
	Evaluation of influence of high energy particles
	Evaluation of influence of target, beam window and void in the beam duct
	Simulation of LBE coolant
Demonstration of controllability of a hybrid system driven by an accelerator	Feedback control of reactor power by beam intensity adjustment
	Investigation of system behaviour at beam trip and re-start
	Evaluation of temperature effect for core and target
	Investigation of instability of system caused by the subcriticality and the annular arrangement of the core
	Determination of energy gain factor
Transmutation performance of MA and LLFP	Measurement of MA transmutation rate
	Measurement of MA and LLFP sample reactivity worth
	Study of moderated region for LLFP transmutation
	Simulation of MA-loaded nitride core
	Measurement of cross section data by TOF technique

Figure 61: Image of Pin-type fuel partial loading section for TEF-P



The distinguished points of TEF-P in comparison with the existing experimental facilities can be summarised as follows: 1) both the high-energy proton beam and the nuclear fuel are available, 2) the maximum neutron source intensity of about 10^{12} n/s is

strong enough to perform precise measurements even in the deep subcritical state (e.g. k_{eff}=0.90), and, is low enough to easily access to the assembly after the irradiation, 3) a wide range of pulse-width (1ns – 0.5ms) can be available by the laser charge exchange technique and, 4) MA and LLFP can be used as a shape of foil, sample and fuel by installing an appropriate shielding and a remote handling devise.

5. Summary and recommendations

The EGIEMAM reviewed the state-of-the-art minor actinides nuclear data, performed uncertainty analyses and target accuracy assessments for minor actinide management and stressed the need to improve nuclear data. Existing integral experiments were also reviewed from the viewpoint of MA management and a lack of experiments and accuracy were identified. Because of proprietary considerations, most of the results are not fully available.

As there is only a limited number of facilities and limited expertise and resources (materials, manpower, funding), the expert group stressed the need for a concerted effort paving the way for a common experimental programme, where resources can be optimised towards improving MA nuclear data knowledge.

5.1 Additional integral experiments

The expert group members discussed the integral experiments for minor actinides and minor isotopes in major actinides (MIs). After the discussion, they recognised the importance of integral experiments not only for MAs and MIs but also for other nuclides including major actinides. Based on current MA nuclear data knowledge and from the viewpoint of MA management, the following integral experiments should be carried out:

- Reaction rate measurements in fast systems:
  - capture rate for ^{237}Np, ^{241}Am, ^{243}Am, ^{244}Cm and ^{245}Cm;
  - fission rate for ^{245}Cm.
- Small sample reactivity worth measurements:
  - necessity to accumulate data for MA in fast system;
  - pre-analysis (e.g. estimation of reactivity worth and sensitivity analysis) is important.
- Irradiation experiments:
  - irradiation experiments in fast systems to complement the other integral experiments;
  - analysis of existing data supported through international co-operation.
- Mock-up experiments:
  - mock-up experiments using massive amounts of MAs will be needed in the future;
  - start of design studies to prepare the experiments (estimation of MA fuel contents, amounts and the effectiveness of large inventory [~kg] experiments for parametric study to simulate design cores).

5.2 Recommendation of an action programme for international co-operation

In addition to the proposal of the integral experiments for MAs, the EGIEMAM recommended the following actions to be performed through international co-operation:

- Comparisons in order to obtain high reliability for integral data and analysis results:
  - international-comparison of the reaction rate measurements at common facilities;
  - international-comparison of chemical analysis results with the same irradiated samples in PIE;
  - comparison of calculation results of benchmark experiments (including the selection of benchmark experiments).
- Co-operation on differential nuclear data measurements complementary to the other integral experiments:
  - making a proposal to WPRS;
  - sharing new cross-section measurements and evaluations in nuclides, reactions and/or neutron energy ranges.
- Co-operation on design study for the specification of MA cores for integral experiments at zero power facility in support of current selected target designs for MA management.

To realise these proposals, the members discussed new actions using examples from past international collaboration related to integral experiments such as the IRMA campaign in 1984 and the international benchmark experiment for effective delayed neutron fraction in 1995. Finally, the EGIEMAM proposed the following actions:

- Joint design of reactor physics MA measurements in selected facilities:
  - selection of reactions and nuclides to be measured including assessment of the new measurement techniques/data, and specifications of the integral experiments including target uncertainties;
  - consideration and determination of facilities and required expertise;
  - development and co-ordination of the experimental programme (including assessment and sharing of resources and results, time schedules, and costs…).
- International collaboration of the irradiation programme for MA:
  - selection of reactions and nuclides to be irradiated including assessments of the new measurement techniques/data, and specifications of the integral experiments including target uncertainties;
  - selection of irradiation facilities;
  - considering preparation of the samples for irradiation and time schedule for irradiation;
  - development and co-ordination of the irradiation programme (including assessments and sharing of resources and results, time schedules and costs…).

In addition, the EGIEMAM recommended that the WPRS and WPFC consider an international comparison of the analysis of integral experiments for MAs in the selection of experiments, the creation of benchmark models, the analysis of benchmark problems and the comparison of analysis results.

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List of integral experiments for minor actinide management

Basic experiments using a critical facility

Facility: VENUS (SCK-CEN, Belgium)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Fission rate ratio (to ^{235}U fission)	Fission	^{237}Np ^{239}Pu ^{240}Pu ^{242}Pu ^{234}U ^{238}U	25 μg 21 μg 367 μg 340 μg 230 μg 102 μg	>99%	May 2011	Fast	SCK•CEN	2.4 - 4.2%.
Assessments	The detailed description of the core configuration as well as the experimental results are available for the FP6 (GUINEVERE) and FP7 (FREYA, ANDES) partners but not published yet.							
Remarks (References etc.)	Analysis is on-going. Some preliminary info is in: A.Kochetkov VENUS-F: A First Fast Lead Critical Core for Benchmarking, in ASTM, Vol. 9, No. 3, January 2012.							

Facility: MASURCA (CEA, France)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Fission rate ratio (to ^{235}U fission)	Fission	^{238}U ^{239}Pu	Some 10 μg	High	1989-1992	F	CEA	
Assessments	Detailed information is not available.							
Remarks (References etc.)								

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Fission rate ratio (to ^{235}U fission)	Fission	^{237}Np ^{238}U ^{239}Pu	Some 10 μg	High	1993-1994	F	CEA	
Assessments	Detailed information was made available to all programme partners as a benchmark.							
Remarks (References etc.)	BERENICE experimental programme (cores for β_{eff} measurement). P. BERTRAND et al., Proc. PHYSOR'96, Mito, Japan, BERENICE – Inter laboratory comparison of β_{eff} measurement techniques at MASURCA.							

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Fission rate ratio (to ^{235}U fission)	Fission	^{237}Np ^{238}U ^{238}Pu ^{239}Pu ^{240}Pu ^{241}Pu ^{242}Pu ^{241}Am ^{243}Am	Some 10 μg	High	1994-1997	F	CEA	
Assessments	Detailed information is proprietary.							
Remarks (References etc.)	CIRANO experimental programme. G. Rimpault et al., Proc. PHYSOR-1998, Long Island, US.							

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Fission rate ratio (to ^{235}U fission)	Fission	^{237}Np ^{238}U ^{238}Pu ^{239}Pu ^{240}Pu ^{241}Pu ^{242}Pu ^{241}Am ^{243}Am	Some 10 μg	High	1998-1999	F	CEA	
Assessments	Detailed information is proprietary.							
Remarks (References etc.)	COSMO experimental programme. G. Rimpault et al, Proc. PHYSOR-2000, Pittsburgh, US: ERANOS neutronic calculations of a $^{11}B_4C$ moderated subassembly and experimental validation in MASURCA.							

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Fission rate ratio (to ^{235}U fission)	Fission	^{232}Th ^{237}Np ^{238}U ^{239}Pu ^{240}Pu ^{242}Pu ^{241}Am ^{243}Am	Some 10 μg	High	2001-2004	F	CEA	
Assessments	A detailed description of the experiments was made available to the MUSE-4 EC FP5 partners as a benchmark.							
Remarks (References etc.)	MUSE-4 experimental programme. J.F. Lebrat et al., Global results from deterministic and stochastic analysis of the MUSE-4 experiments on the neutronics of accelerator-driven systems, Nucl. Sci. Eng. 158 (2008).							

Facility: EOLE (CEA, France)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Fission rate ratio (fission Chambers)	Fission	^{237}Np ^{239}Pu	352 µg 116 µg	100% 98.5%	2005 2005	T		
Assessments	Detailed information is proprietary.							
Remarks (References etc.)	Advanced High-burn up 100% MOX ABWR Spectrum (FUBILA Programme). P. Blaise, N. Huot, N. Thiollay, P. Fougeras, A. Santamarina.High-burn up 10x10 100% MOX ABWR core physics analysis with APOLLO2.8 and TRIPOLI-4.5 Codes. Annals of Nuclear Energy 37 (2010) 889-909. 4 May 2010.							

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Fission rate ratio (fission Chambers)	Fission	^{237}Np ^{239}Pu ^{240}Pu ^{241}Pu	352 µg 116 µg 195 µg 100 µg	100% 98.5% 98.4% 93.4%	1996 1996 1996 1996	T		
Assessments	Detailed information is proprietary.							
Remarks (References etc.)	Over-moderated UOX PWR Spectrum (MISTRAL Programme).							

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Fission rate ratio (fission Chambers)	Fission	^{237}Np ^{238}Pu ^{239}Pu ^{240}Pu ^{241}Pu ^{242}Pu	352 µg 130 µg 116 µg 195 µg 100 µg 142 µg	100% 71.1% 98,5% 98.4% 93.4% 94.8%	1997 - 1998 1997 1997 - 1998 1997 1997 - 1998 1997	T		
Assessments	Detailed information is proprietary.							
Remarks (References etc.)	Over-moderated MOX PWR Spectrum (MISTRAL Programme). S. Cathalau, P. Fougeras, P. Blaise, N. ThiollaY, A. Santamarina, O. Litaize, C. Chabert, T. Yamamoto, Y. Iwata, T.Umano, R. Kanda, P. Girieud, D. Biron, M. Tatsumi, T. Kan, Y. Ando, K. Ishii. Full MOX recycling in ALWR: Lessons Drawn through the MISTRAL Programme. PHYSOR 2002, Seoul, Republic of Korea October 7-10, 2002.							

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Fission rate ratio (fission Chambers)	Fission	^{238}Pu ^{239}Pu ^{240}Pu ^{241}Pu ^{242}Pu	31.7 µg 116 µg 195 µg 100 µg 112 µg	83 % 99.1% 98.4% 93.4% 98.7%	1993 1993 1993 1993 1993	T		
Assessments	Detailed information is proprietary.							
Remarks (References etc.)	MOX PWR Spectrum (EPICURE Programme). J.P. Chauvin, J.C. Cabrillat, P. Fougeras, S. Cathalau, P.J. Finck. "EPICURE: an Experimental Program Devoted to the Validation of the Calculation Scheme for the Pressurised Water Reactor Recycling MOX Fuel". Int. Conf on the Physics of Reactors PHYSOR'96 pg E-133. MITO, Japan, September 16-20, 1996.							

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Fission rate ratio (fission Chambers)	Fission	^{239}Pu ^{240}Pu ^{241}Pu ^{242}Pu	116 µg 195 µg 100 µg 112 µg	99.1% 98.4% 93.4% 98.7%	1987 1987 1987 1987	I		
Assessments	Detailed information is proprietary.							
Remarks (References etc.)	MOX HCLWR Spectrum (ERASME Programme). L. Martin-Deidier, A. Santamarina, S. Cathalau, JM. Gomit, J.P. Chauvin. ERASME: an extensive experiment for LWHCR design qualification. Topical meeting on reactor Physics. September 1986. Saratoga, United States.							

Facility: MINERVE (CEA, France)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Oscillations of separated MA isotope samples	Capture + fission (reactivity measure ment)	^{237}Np id ^{238}Pu ^{239}Pu ^{240}Pu ^{241}Pu ^{242}Pu ^{241}Am id ^{243}Am ^{244}Cm Id ^{232}Th	0.10 g 0.58 g 0.37 g 0.57 g 0.15 g 0.18 g 0.48 g 0.06 g 0.18 g 0.52 g 0.64 g 0.48 g 1.86 g	100% 100% 100% 100% 100% 100% 100% 100% 100% 100% 100% 100% 100%	2005-2012	Mostly T, some I	CEA & ORNL	
Assessments	The details of the experiments are restricted to the programme partners.							
Remarks (References etc.)	UOX PWR Spectrum, MOX PWR Spectrum, Over-moderated UO2 spectrum, Epithermal spectrum. In the OSMOSE programme. JP. Hudelot, R. Klann, M. Antony, S. Testaniere, P. Fougeras, F. Jorion, N. Drin, L. Donnet, C. Leorier. The OSMOSE Program for the Qualification of Integral Cross Sections of Actinides: Preliminary Experimental Results in a PWR-UOx Spectrum, PHYSOR 2006 Advances in Nuclear Analysis and simulation. September 10-14 2006 Vancouver Canada. See also D. BERNARD et al. ND2010 Conf. Proceedings, Republic of Korea, 2010.							

Facility: FCA (JAEA, Japan)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Fission rate ratio	Fission	^{237}Np ^{238}Pu ^{239}Pu ^{242}Pu ^{241}Am ^{243}Am ^{244}Cm	40-120 µg	78-100 %	1981-1982	F	JAERI	~2% (~4% for ^{244}Cm/^{239}Pu)
Assessments	Summary for these experiments are available from the references.							
Remarks (References etc.)	FCA-IX cores, Mukaiyama T. et al.; Proc.of the Int.Conf. Nuclear Data for Basic and Applied Science, (Santa Fe, US, 1985) Gordon and Breach Science Publishers, pp.483-488. S. Okajima et al.; Proc. of PHYSOR 2008.							

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Sample reactivity worth	Capture	^{237}Np ^{238}Pu ^{240}Pu ^{241}Am ^{243}Am	15-20 g		1981-1982	F	ORNL	~2 pcm
Assessments	Summary for these experiments are available from the references.							
Remarks (References etc.)	FCA-IX cores, Mukaiyama T. et al.; Proc.of the Int.Conf. Nuclear Data for Basic and Applied Science, (Santa Fe, US, 1985) Gordon and Breach Science Publishers, pp.483-488. S. Okajima et al.; Proc. of PHYSOR 2008.							

Facility: TCA (JAEA, Japan)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Sample reactivity worth	Capture	^{237}Np ^{241}Am	15-20 g	99% 90 %	2006	T	ORNL	1.3-2.4 %
Assessments	Detailed information for these experiments, including geometric and material data are available from the reference.							
Remarks (References etc.)	T. Sakurai et al., J. Nucl. Sci. Technol., 46, 624 (2009).							

Facility: KUCA (Kyoto Univ., Japan)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Fission rate ratio	Fission	^{237}Np ^{241}Am	~ 80 µg ~ 8 µg	>99% >99%	1993 1997	T T	IAEA IAEA	2.0-2.3 % 2.1-2.3 %
Assessments	Detailed information for these experiments, including geometric and material data are available from the references.							
Remarks (References etc.)	H. Unesaki et al., J. Nucl. Sci. Technol., 37, 627 (2000). H. Unesaki et al., J. Nucl. Sci. Technol., 38, 600 (2001).							

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Capture rate ratio	Capture	^{237}Np	~1 mg	>99%	1999	T	JAERI	3.4-4.0%
Assessments								
Remarks (References etc.)	T. Iwasaki et al., Nucl. Sci. Eng., 136, 321 (2000).							

Facility: BFS-60, -71, -73*, -75*, -61**, -77**, -87** (IPPE, Russian Federation)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Fission rate ratio	Fission	^{237}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{244}Cm	μg 207 30 20-30 60-110 30 200 300 410 70	98 88 95-99 98 99 95 99 94 87	1987-2000	Fast	IPPE/ SCK•CEN	1.6-4.6%
Assessments	Detailed information for these experiments, are available from the references, however geometric and material data are the property of IPPE and CEA, KAERI, JAEA except BFS-73. The detailed information of BFS-73 geometric and material data is in IRPhEP data base.							
Remarks (References etc.)	1.A. Kochetkov, et al. Study of Minor Actinides Data at BFS-73-1Fast Critical Assembly with Metal Fuel. Proceedings of ICAPP '05, Seoul, Republic of Korea, May 15-19, 2005. 2.A. L.Kochetkov,Y.Khomyakov et al. BFS critical experiments for the Minor Actinides data correction, PHYSOR-2006, Vancouver, Canada, 2006.							

Facility: BFS-73

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Sample reactivity worth		^{237}Np, ^{241}Am	g 9-22 7-9	99 99	1989			^{237}Np: -0.083±0.014 ^{241}Am: -0.083±0.016
Assessments								
Remarks (References etc.)	IRPhEP data base: "BFS-73-1: Experimental model of sodium-cooled fast reactor with core of metal uranium fuel 18.5% enrichment and depleted uranium dioxide blanket".							

**Critical assemblies with sodium coolant and metal fuel*

***Critical assemblies with Pb and Pb-Bi coolant and nitride fuel*

Facility: BFS BFS-69, BFS-71, BFS-73 and BFS-77 (IPPE, Russian Federation)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Capture to Fission (^{239}Pu) Rate ratio	Capture	^{237}Np			1985-1987	F	IPPE/ SCK•CEN	3.2 – 4.6 %
Assessments	Detailed information for these experiments, are available from the references, however geometric and material data are the property of IPPE and CEA.							
Remarks (References etc.)	V.Doulin, GLOBAL-97 GLOBAL-99. V.Doulin, Measurement of the ratio of the average ^{237}Np capture cross-section to the ^{239}Pu fission cross-section in two BFS critical assemblies, Atomic Energy, vol. 84, № 6 , p.407, 1998.							

Facility: PROTEUS (PSI, Switzerland)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Reaction rate ratios	(n,f), (n, γ)	^{237}Np	Deposit 180 μg/cm^2 on Al. backing		~1975	F and I	AERE Harwell	
Assessments	Spectral indices have been measured relative to fission in ^{239}Pu and typical experimental uncertainties are about 2%. The values have been used recently (Physor'12 and Wonder'12 to demonstrate limitation of the recent nuclear data libraries using a new MCNPX model of the facility).							
Remarks (References etc.)	Gas-Cooled Fast Reactor ANE Vol.6. pp.585-589 + Internal docs. G. Perret et al., "Re-analysis of the gas-cooled fast reactor experiments at the zero power facility Proteus – spectral indices" Physor'12.							

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Reaction rate ratios	(n,f)	^{237}Np	48 μg/cm^2 on Al. backing		~1980	I	AERE Harwell	
Assessments								
Remarks (References etc.)	Light water high conversion internal docs.							

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Reaction rate ratios	(n,f)	^{241}Pu	Fission chamber		~1980	I	AERE Harwell	
Assessments	Uncertainty on F1/F9 is about 5%.							
Remarks (References etc.)								

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Reaction rate ratios	(n,γ)	^{237}Np, ^{242}Pu	0.2 mg/cm^2 deposit on Al backing		~1980	I	AERE Harwell	
Assessments	Uncertainty on C2/F9 is about 3%.							
Remarks (References etc.)								

Facility: ^{240}Pu JEZEBEL (LANL, United States)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Replacement measurement of material reactivity worth	(n,f), (n,γ), ν	^{244}Cm	2.6128 g	95.5%	1964	F		~5%
Assessments	Detailed information for these experiments, including geometric and material data, is available from the SPEC-MET-FAST-001 reference.							
Remarks (References etc.)	1. SPEC-MET-FAST-001, ICSBEP Handbook, NEA/NSC/DOC(95)03/VII (2010). 2. D. M. Barton, "Central Reactivity Contributions of ^{244}Cm, ^{239}Pu, and ^{235}U in a Bare Critical Assembly of Plutonium Metal", Nucl. Sci. and Eng., 33, 51-55 (1968).							

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Replacement measurement of material reactivity worth	(n,f), (n,γ), ν	^{239}Pu	2.7780 g	95.81%	1964	F		~3%
Assessments	Detailed information for these experiments, including geometric and material data, is available from the SPEC-MET-FAST-001 reference.							
Remarks (References etc.)	1. SPEC-MET-FAST-001, ICSBEP Handbook, NEA/NSC/DOC(95)03/VII (2010). 2. D.M. Barton, "Central Reactivity Contributions of ^{244}Cm, ^{239}Pu, and ^{235}U in a Bare Critical Assembly of Plutonium Metal", Nucl. Sci. and Eng., 33, 51-55 (1968).							

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Replacement measurement of material reactivity worth	(n,f), (n,γ), ν	^{235}U	3.1633 g	93.2%	1964	F		~2%
Assessments	Detailed information for these experiments, including geometric and material data, is available from the SPEC-MET-FAST-001 reference.							
Remarks (References etc.)	1. SPEC-MET-FAST-001, ICSBEP Handbook, NEA/NSC/DOC(95)03/VII (2010). 2. D.M. Barton, "Central Reactivity Contributions of ^{244}Cm, ^{239}Pu, and ^{235}U in a Bare Critical Assembly of Plutonium Metal", Nucl. Sci. and Eng., 33, 51-55 (1968).							

Facility: ^{240}Pu JEZEBEL (LANL, United States)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Replacement measurement of material reactivity worth	(n,f), (n,γ), ν	^{238}Pu	12.410 g	79.65%	1967-1968	F		~5%
Assessments	Detailed information for these experiments, including geometric and material data, is available from the SPEC-MET-FAST-002 reference.							
Remarks (References etc.)	1. SPEC-MET-FAST-002, ICSBEP Handbook, NEA/NSC/DOC(95)03/VII (2010). 2. W.F. Stubbins, D. M. Barton, and F. D. Lonadier, "The Neutron Production Cross-Section of ^{238}Pu in a Fast Spectrum", Nucl. Sci. and Eng., 25, 377-382 (1966).							

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Replacement measurement of material reactivity worth	(n,f), (n,γ), ν	^{239}Pu	11.0369 g	94.69%	1967-1968	F		~3%
Assessments	Detailed information for these experiments, including geometric and material data, is available from the SPEC-MET-FAST-002 reference.							
Remarks (References etc.)	1. SPEC-MET-FAST-002, ICSBEP Handbook, NEA/NSC/DOC(95)03/VII (2010). 2. W.F. Stubbins, D. M. Barton, and F. D. Lonadier, "The Neutron Production Cross Section of ^{238}Pu in a Fast Spectrum", Nucl. Sci. and Eng., 25, 377-382 (1966).							

Facility: FLATTOP (LANL, United States)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Replacement measurement of material reactivity worth	(n,f), (n,γ), ν	^{237}Np	28.393 g	99.87%	1994-1995	F		~6%
Assessments	Detailed information for these experiments, including geometric and material data, is available from the SPEC-MET-FAST-003 reference.							
Remarks (References etc.)	1. SPEC-MET-FAST-003, ICSBEP Handbook, NEA/NSC/DOC(95)03/VII (2010). 2. R.G. Sanchez, "Critical Mass of Np-237", Proceedings of the Fifth International Conference on Nuclear Criticality Safety, Albuquerque, NM, September 17-21, 1995, 6, 182-189 (1995).							

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Replacement measurement of material reactivity worth	(n,f), (n,γ), ν	^{235}U	29.909 g	93.2%	1994-1995	F		~6%
Assessments	Detailed information for these experiments, including geometric and material data, is available from the SPEC-MET-FAST-003 reference.							
Remarks (References etc.)	1. SPEC-MET-FAST-003, ICSBEP Handbook, NEA/NSC/DOC(95)03/VII (2010). 2. R.G. Sanchez, "Critical Mass of Np-237", Proceedings of the Fifth International Conference on Nuclear Criticality Safety, Albuquerque, NM, September 17-21, 1995, 6, 182-189 (1995).							

Facility: Comet (LANL, United States)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Criticality	(n,f), (n,γ), ν	^{242}Pu, ^{239}Pu, and ^{235}U	kgs	High	1979	F		Range from ~500 to 1 200 pcm
Assessments	Detailed information for these experiments, including geometric and material data, is available from the SPEC-MET-FAST-004 reference. Seven critical configurations with varying sensitivity to ^{242}Pu were measured with uncertainties in k_{eff} ranging from 500 to 1 200 pcm.							
Remarks (References etc.)	1. SPEC-MET-FAST-004, ICSBEP Handbook, NEA/NSC/DOC(95)03/VII (2010).							

Facility: Planet (LANL, United States)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Criticality	(n,f), (n,γ), ν	^{237}Np and ^{235}U	6.0704 kg Np and 62.555 kg HEU	Np: 98.8% Np (100% ^{237}Np); 0.0344% U; 0.0366% Pu; 0.1829% Am	2002	F		340 pcm
Assessments	Detailed information for these experiments, including geometric and material data, is available from the SPEC-MET-FAST-008 reference. Critical configuration uncertainty in k_{eff} = 340 pcm. Np sphere surrounded by HEU shells. Note: Configuration has high sensitivity to HEU.							
Remarks (References etc.)	1. SPEC-MET-FAST-008, ICSBEP Handbook, NEA/NSC/DOC(95)03/VII (2010). 2. R. Sanchez et al., "Criticality of a ^{237}Np Sphere," Nucl. Sci. and Eng., 158, 1-14 (2008).							

Facility: Planet (LANL, United States)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Criticality	(n,f), (n,γ), ν	^{237}Np and ^{235}U	6.0704-kg Np and 26.9645 kg HEU	Np: 98.8% Np (100% ^{237}Np); 0.0344% U; 0.0366% Pu; 0.1829% Am	2004	F		290 pcm
Assessments	Detailed information for these experiments, including geometric and material data, is available from the SPEC-MET-FAST-011 reference. Critical configuration uncertainty in k_{eff} = 290 pcm. Np sphere surrounded by HEU shells and reflected by polyethylene. Note. Configuration has high sensitivity to HEU and polyethylene.							
Remarks (References etc.)	1. SPEC-MET-FAST-011, ICSBEP Handbook, NEA/NSC/DOC(95)03/VII (2010).							

Facility: Planet (LANL, United States)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Criticality	(n,f), (n,γ), ν	^{237}Np and ^{235}U	6.0704-kg Np and 34.323 kg HEU	Np: 98.8% Np (100% ^{237}Np); 0.0344% U; 0.0366% Pu; 0.1829% Am	2003-2004	F		390 pcm
Assessments	Detailed information for these experiments, including geometric & material data, is available from the SPEC-MET-FAST-014 reference. Critical configuration uncertainties in k_{eff} = 390 pcm. Np sphere surrounded by HEU shells and reflected by low-carbon steel. Note: Configuration has high sensitivity to HEU and iron.							
Remarks (References etc.)	1. SPEC-MET-FAST-014, ICSBEP Handbook, NEA/NSC/DOC(95)03/VII (2010).							

Facility: GODIVA, JEZEBEL, JEZEBEL-23, FLATTOP-25, FLATTOP-Pu, FLATTOP-23 (LANL, United States)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Reaction rates and spectral ratios	f^{37}/f^{25} c^{51}/f^{49}	^{237}Np and ^{241}Am	various	High	various	F		~5%
Assessments	Experimental values are given in NDS references; however, physical description of experiments is limited. Uncertainties are not well evaluated, but considered to be 5%.							
Remarks (References etc.)	1. G. Hansen, private communication to R. MacFarlane (1984). 2. D.W. Barr, private communication to M. B. Chadwick. 3. Nuclear Data Sheets 112 (2011) 2997–3036. 4. Nuclear Data Sheets 107 (2006) 2931–3060.							

Facility: ZEBRA (UKAEA, United Kingdom)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties (1 σ)
Fission rate ratio (to ^{239}Pu fission)	Fission	^{238}U ^{240}Pu ^{241}Pu ^{242}Pu ^{241}Am ^{243}Am ^{244}Cm	1839 μg 1602 μg 202 μg 680 μg 9.21 μg 46.33 μg 1.38 μg	>99%	1976	Fast	UKAEA	1.2% 2.3% 3.1% 3.1% 0.9% 0.9% 6.1%
Assessments	Detailed technical reports are available for these experiments, including geometric and material data and a breakdown of the main contributors to the measurement uncertainty. It is considered that this experiment offers very useful information for validation of MA.							
Remarks (References etc.)	AEEW-R1090 Actinide Fission rate Measurements in ZEBRA –D W Sweet Jan1977.							

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties (1 σ)
Capture rate ratio (to ^{239}Pu fission)	Capture	^{241}Am ^{243}Am	3 ^{241}Am samples: 9.516, 9.138 and 9.974 mg. 3 ^{243}Am samples: ~20 mg each	97.3$^{w}/_{o}$	08/09/1972	Fast	UKAEA	3.1%
Assessments	A detailed technical report is available for the Zebra Core 12 ^{241}Am experiment, including geometric & material data and a breakdown of the main contributors to the measurement uncertainty. It is considered that these experiments offer very useful information for validation of MA.							
Remarks (References etc.)	Two ^{241}Am experiments were performed, in Zebra Cores 12 and 14, respectively. Two ^{243}Am experiments were performed, in Zebra Cores 14 and 16, respectively. Only a conference summary paper, giving the absolute measured capture cross-sections (derived from the measured reaction rate ratios using assumed ^{239}Pu cross-sections), is available for the Core 14 ^{241}Am experiment and the two ^{243}Am experiments. AEEW-R7363 The cross-section for the Production of ^{242}Cm from ^{241}Am in a Fast Reactor–D.W Sweet et al, January 1973. IAEA-SM-244/38, Some Aspects of Fast Reactor Operation Studied in ZEBRA, J.E Sanders et al (date unknown).							

Sample irradiation experiments

Facility: TRIGA RC-1 (ENEA, Italy)

Meas. items	Reaction	Nuclide	Mass of material	Purity	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Capture rate	Capture	^{241}Am	4 samples: 0.299 up to 0.460 mg	100%	2004-2005	Thermal/ Low epithermal region	ENEA	Range from: (1.4 to 24)%. Bare samples (2.4 to 23)%. Cd-lined samples
Assessments	Feasibility and validation of ^{242m}Am build-up model have been demonstrated. A publication and fragmented technical information are available. Discrepancies and inconsistencies suggest: re-evaluation of the experiment will be right and opportune.							
Remarks (References etc.)	Production of ^{242m}Am, P. Benetti et al., Nuclear Instruments and Methods in Physics Research A 564 (2006) 482-485.							

Facility: PFR (PFR, United Kingdom)

Core	Nuclide	Mass of material	Purity	Irradiation period	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
PFR	$^{230, 232}$Th, ^{231}Pa $^{233, 234, 235, 236, 238}$U, ^{237}Np $^{238, 239, 240, 241, 242, 244}$Pu-$^{241, 243}$Am $^{243, 244, 246, 248}$Cm	0.4 – 12 mg	60-99 (%)	492 EFPD	1982-1988	F	ORNL	< 0.3 % for U and Pu (<1.2% for ^{238}Pu) 3-4% for Np 0.7-3.5% for Am 0.6-1.0% for Cm
Assessments	Detailed information including sample preparation, isotopic composition before and after irradiation, and irradiation history are available. However, detailed information for core material is not available.							
Remarks (References etc.)	N.Shinohara et al., Nucl. Sci. and Eng., 144(2) pp.115-128 (2003). K.Tsujimoto et al., Nucl. Sci. and Eng, 144(2) pp.129-141 (2003).							

Facility: PHENIX (CEA, France)

Core	Nuclide	Mass of material	Purity	Irradiation period	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
PHENIX FBR core	^{232}Th, $^{233, 234, 235, 238}$U, ^{237}Np $^{238, 239, 240, 241, 242}$Pu Am-241, -243 ^{244}Cm	Typically 10-20 mg	High (separated isotopes)	1 : 1974-1975, 179 EFPD 2 : 1979-1980, 316 EFPDs.	1976-1987	F	CEA	
Assessments								
Remarks (References etc.)	PROFIL-1 and PROFIL-2 Feedback to capture integral nuclear data, J. Tommasi and G. Noguère, Analysis of the PROFIL and PROFIL-2 sample irradiation experiments in Phénix for JEFF-3.1 nuclear data validation, Nucl. Sci. Eng. 160 (2008).							

Core	Nuclide	Mass of material	Purity	Irradiation period	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
PHENIX FBR core	^{237}Np and ^{241}Am transmutation	2% to 45% in standard fuel	High (single Np and Am isotopes)	1986-1988, 382 EFPD	1989-1992	F	CEA & ITU	
Assessments								
Remarks (References etc.)	SUPERFACT experiment, C. Prunier et al., Some specific aspects of homogeneous americium-and neptunium-based fuels transmutation through the outcomes of the SUPERFACT experiment in phenix fast reactor, Nucl. Tech.119 (1997).							

Core	Nuclide	Mass of material	Purity	Irradiation period	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
PHENIX FBR core	^{232}Th, $^{233, 234, 235, 238}$U, ^{237}Np $^{238, 239, 240, 242}$Pu $^{241, 243}$Am ^{244}Cm	Typically a few mg	High (separated isotopes)	R: 2004-2005, 252 EFPD M: 2006-2008, 227 EFPDs.	PIEs started in 2008, will take several years to be complete d	F (R) and softene d (M)	CEA	
Assessments	PIE to be done over 2010-2015.							
Remarks (References etc.)	PROFIL-R and PROFIL-M. G. Gaillard-Groléas, F. Sudreau and D. Warin, Proc. GLOBAL'2003, New orleans, US : Phénix irradiation program on fuels and targets for transmutation. http://www.nea.fr/html/pt/docs/iem/lasvegas04/10_Session_IV/S4_01.pdf.							

Facility: PWRs (EDF, France)

Core	Nuclide	Mass of material	Purity	Irradiation period	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
UOx and MOx cores	As measured in PIEs	Fuel segments	Mixture	Several cycles	Date of PIEs	T		
Assessments								
Remarks (References etc.)	OECD/NEA JEFF-3.1.1 Report 22							

Facility: JOYO (JAEA, Japan)

Core	Nuclide	Mass of material	Purity	Irradiation period	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
JOYO MK-II core	 ^{237}Np, ^{241}Am, ^{243}Am, ^{244}Cm	μg ~ 200 , ~ 100 ~ 100 ~ 100	 100% 100% 88% 94%	250-280 EFPD	1994-1999	F	Frace	~ 20% for FIMA on Am samples
Assessments	Detailed information for these experiments are available from the references.							
Remarks (References etc.)	S.Koyama and T.Mitsugashira, J. Nucl. Sci. Eng., Suppl. 5, pp.55-64 (2008). K. Sugino, Proc. of International Conference on the Advanced Nuclear Fuel Cycles and Systems GLOBAL2007, Boise, ID (2007).							

Facility: BN-350 (Russian Federation, Kazakhstan)

Core	Nuclide	Mass of material	Purity	Irradiation period	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
LEZ - Cell 89	^{241}Am ^{237}Np ^{238}Pu	142 109 110	99.9% 100% 80%	1990-1993	1998 1998 1998	Fast	IPPE	
Assessments	Detailed information including sample preparation, isotopic composition before and after irradiation, and irradiation history are available. Detailed information for core material is the property of IPPE and BN-350.							
Remarks (References etc.)	1. Evaluation of the activation and burn-up experiments carried out in BN-350 reactor, Yu.S. Khomyakov et al. PHYSOR-02, Republic of Korea, October 2002. 2. Calculation and experimental studies on minor actinides samples irradiation in fast reactors. A. Kochetkov et al. CD-ROM Proc. of 7th Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation, 14-16 October 2002, Jeju (Republic of Korea), OECD, Paris, 2003.							

Core	Nuclide	Mass of material	Purity	Irradiation period	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
LEZ - Cell110	^{240}Pu ^{234}U	150 76	78%	1990-1993	1999 1999	Fast	IPPE	
Assessments	Detailed information including sample preparation, isotopic composition before and after irradiation, and irradiation history are available. Detailed information for core material is the property of IPPE and BN-350.							
Remarks (References etc.)	1. Evaluation of the activation and burn-up experiments carried out in BN-350 reactor, Yu.S. Khomyakov et al. PHYSOR-02, Republic of Korea, October 2002. 2. Calculation and experimental studies on minor actinides samples irradiation in fast reactors. A. Kochetkov et al. CD-ROM Proc. of 7th Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation, 14-16 October 2002, Jeju (Republic of Korea), OECD, Paris, 2003.							

Core	Nuclide	Mass of material	Purity	Irradiation period	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
LEZ - Cell 243	^{241}Am ^{240}Pu ^{244}Cm ^{244}Cm	146 16 4 4	99.9% 78% 65% 65%	1991-1993	1999 2000 2000 2000	Fast	IPPE	
Assessments	Detailed information including sample preparation, isotopic composition before and after irradiation, and irradiation history are available. Detailed information for core material is the property of IPPE and BN-350.							
Remarks (References etc.)	1. Evaluation of the activation and burn-up experiments carried out in BN-350 reactor, Yu.S. Khomyakov et al. PHYSOR-02, Republic of Korea, October 2002. 2. Calculation and experimental studies on minor actinides samples irradiation in fast reactors. A. Kochetkov et al. CD-ROM Proc. of 7th Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation, 14-16 October 2002, Jeju (Republic of Korea), OECD, Paris, 2003.							

Core	Nuclide	Mass of material	Purity	Irradiation period	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Uranium low enrichment core	^{235}U ^{239}Pu MOX fuel	300-480	88% 95% 74%	145 EFPD 1981	1984	Fast	IPPE	
Assessments	Detailed information including sample preparation, isotopic composition before and after irradiation, and irradiation history are available. Detailed information for core material is the property of IPPE and BN-350.							
Remarks (References etc.)	1. Evaluation of the activation and burn-up experiments carried out in BN-350 reactor, Yu.S. Khomyakov et al. PHYSOR-02, Republic of Korea, Oct. 2002. 2. Calculation and experimental studies on minor actinides samples irradiation in fast reactors. A. Kochetkov et al. CD-ROM Proc. of 7th Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation, 14-16 October 2002, Jeju (Republic of Korea), OECD, Paris, 2003.							

Core	Nuclide	Mass of material	Purity	Irradiation period	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
MOX Core	MOX fuel	50-150	(21% Pu, 79 % U)	1982-1983	1984-1986	Fast	IPPE	
Assessments	Detailed information including sample preparation, isotopic composition before and after irradiation, and irradiation history are available. Detailed information for core material is the property of IPPE and BN-350.							
Remarks (References etc.)	1. Evaluation of the activation and burn-up experiments carried out in BN-350 reactor, Yu.S. Khomyakov et al. PHYSOR-02, Korea, Oct. 2002. 2. Calculation and experimental studies on minor actinides samples irradiation in fast reactors. A. Kochetkov et al. CD-ROM Proc. of 7th Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation, 14-16 October 2002, Jeju (Republic of Korea), OECD, Paris, 2003.							

Core	Nuclide	Mass of material	Purity	Irradiation period	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
Blanket	^{232}Th	50-100	100%	1. 81 EFPD 1987-1988 2. 233 EFPD 1990-1992	1990	Fast	IPPE	
Assessments	Detailed information including sample preparation, isotopic composition before and after irradiation, and irradiation history are available. Detailed information for core material is the property of IPPE and BN-350.							
Remarks (References etc.)	1. Evaluation of the activation and burn-up experiments carried out in BN-350 reactor, Yu.S. Khomyakov et al. PHYSOR-02, Korea, Oct. 2002. 2. Calculation and experimental studies on minor actinides samples irradiation in fast reactors. A. Kochetkov et al. CD-ROM Proc. of 7th Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation, 14-16 October 2002, Jeju (Republic of Korea), OECD, Paris, 2003.							

Facility: BOR-60 (Russian Federation)

Core	Nuclide	Mass of material, μg	Purity, %	Irradiation period	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
BOR-60	^{233}Th	419	^{232}Th – 100	1998-2001	2002	Fast	RIAR	
	^{237}Np	408	^{237}Np – 100					
	^{239}Pu	115	^{238}Pu – 0.36 ^{239}Pu – 94.80 ^{240}Pu – 4.66 ^{241}Pu – 0.18 ^{242}Pu – <0.01					
	^{240}Pu	126 a few mg	^{238}Pu – 0.18 ^{239}Pu – 1.00 ^{240}Pu – 98.82					
	^{242}Pu	127	^{242}Pu – 99.54					
	^{241}Am	129	^{241}Am – 100					
	^{243}Am	113	^{241}Am – 2.37 ^{243}Am– 97.62 ^{244}Cm– <0.01					
	^{244}Cm	129	^{244}Cm – 100					
Assessments	Detailed information including sample preparation, isotopic composition before and after irradiation, and irradiation history are available. Detailed information for core material is the property of RIAR (Russian Federation).							
Remarks (References etc.)	Calculation and experimental studies on minor actinides samples irradiation in fast reactors. A. Kochetkov et al. CD-ROM Proc. of 7th Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation, 14-16 October 2002, Jeju (Republic of Korea), OECD, Paris, 2003.							

Facility: ATR (INL, United States)

Core	Nuclide	Mass of material	Purity	Irradiation period	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
ATR	^{232}Th, $^{235, 238}$U, ^{237}Np, $^{238, 239, 240, 241, 242, 244}$Pu-$^{241, 243}$Am $^{244, 248}$Cm	~1 mg	High	FPD February 2012 to September 2013	2012	(F, I)	Internal INL inventory of heavy nuclide materials	
Assessments	Experimental results are not available yet.							
Remarks (References etc.)	MANTRA: Minor Actinide Neutron Transmutation Rate Measurements with Accelerator mass spectrometry. Irradiation to be carried out this year. Filters are applied in order to simulate intermediate and fast spectrum. G. Youinou et al., "An integral reactor physics experiment to infer actinide capture cross-sections from thorium to californium with accelerator mass spectroscop", ND2010, International Conference on Nuclear Data for Science and Technology 2010, Jeju, Republic of Korea, April 2010.							

Facility: EBR-II (ANL, United States)

Core	Nuclide	Mass of material	Purity	Irradiation period	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
EBR-II Mk-III core: S/A X501	U–20.3%Pu–10.0%Zr–2.1%Am–1.3%Np	Two Mk-III fuel elements in experimental S/A X501		339 EFPD	1993 - present	F		
Assessments	Information to model the transmutation in these experiments is available, but largely in internal ANL reports.							
Remarks (References etc.)	M.K. Meyer et al., "The EBR-II X501 minor actinide burning experiment," Journal of Nuclear Materials 392, 176–183 (2009). See also: INL/EXT-08-13835, Note: Final analytical chemistry measurements have not yet been performed.							

Core	Nuclide	Mass of material	Purity	Irradiation period	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
EBR-II Mk-III core: S/A X502, X503	Actinide Dosimetry Measurements	Actinide dosimeters		339 EFPD	1993 - present	F		
Assessments	Information to model the transmutation in these experiments is available, but largely in internal ANL reports.							
Remarks (References etc.)	Various internal ANL reports.							

Mock-up experiments

Facility: BFS (IPPE, Russian Federation)

Core	Fuel	Mass of fuel	Meas. items	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
BFS-67, 69 assemblies	^{237}Np	11 kg of NpO_2	Criticality Reactivity SVRE Cross-section ratios	1994-1996	Fast	IPPE/ SCK•CEN	
Assessments	Detailed information for these experiments are available from the reference. However, geometric and material data are the property of CEA and IPPE.						
Remarks (References etc.)	S.P. Belov et al. "Investigation MA transmutation problem in benchmark experiments at BFS facility with Np in fuel composition." PHYSOR-96, v. IV, p. m-82, 1996, Mito, Japan. BFS-67 assemblies Kotchetkov et al. PHYSOR-2006.						

Core	Fuel	Mass of fuel	Meas. items	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
BFS-66-2	^{237}Np	11 kg of NpO_2	Reactivity SVRE Cross-section ratios	2000-2001	F	IPPE/ SCK•CEN	
Assessments	Detailed information for these experiments as well as geometric and material data are the property of JNC-IPPE.						
Remarks (References etc.)	JNC-IPPE Contract.						

Accelerator-reactor experiments

Facility: VENUS-F – GENEPI3C (SCK•CEN, Belgium)

Programme name	External neutron source	Fuel of Core	Meas. items	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
GUINEVERE	D-T	U	Apparent reactivity, Dynamic measurement	2011- on-going	F	SCK•CEN	
Assessments	Comparison of the PNS area and K_p methods with reference MSM method of subcriticality estimation. The real results are available in the reference.						
Remarks (References etc.)	Coupling in October 2011. H.E. Thyébault et al. "The GUINEVERE Experiment: First PNS Measurements in a Lead Moderated Sub-Critical Fast Core" In Proceedings of ICAPP '12, Chicago, US, June 24-28, 2012.						

Facility: MASURCA (CEA, France)

Programme name	External neutron source	Fuel of Core	Meas. items	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
MUSE-1 MUSE-2 MUSE-3 MUSE-4	^{252}Cf, D-T		Apparent reactivity, Dynamic measurement	1995-2004	F		
Assessments	A detailed description of the MUSE-4 experiments was made available to the MUSE-4 EC FP5 partners as a benchmark.						
Remarks (References etc.)	R. Soule et al., Neutronic Studies in support to ADS: The MUSE Experiment in the MASURCA Facility. Nuclear Science and Engineering, Vol 148 (2004) 124-152. F. Mellier et al., MUSE Final Report, EC FP5 project, June 2005.						

Facility: KUCA (Kyoto Univ., Japan)

Programme name	External neutron source	Fuel of Core	Meas. items	Meas. Date	Energy (F, I, T)	Supplier of materials	Uncertainties
KART	D-T 150 MeV proton + W target	U-Al	Criticality, reaction rate distribution, neutron spectrum, subcriticality, kinetic measurement	2004-2007	T		
Assessments	Detailed information for these experiments, including geometric and material data are available from the references.						
Remarks (References etc.)	C.H. Pyeon et al., *J. Nucl. Sci. Technol.*, 44, 1368 (2007). C.H. Pyeon et al., *J. Nucl. Sci. Technol.*, 45, 1171 (2008). C.H. Pyeon et al., *J. Nucl. Sci. Technol.*, 46, 965 (2009). H. Shahbunder et al., *Ann. Nucl. Energy*, (2010).						

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Review of Integral Experiments for Minor Actinide Management

Spent nuclear fuel contains minor actinides (MAs) such as neptunium, americium and curium, which require careful management. This becomes even more important when mixed oxide (MOX) fuel is being used on a large scale since more MAs will accumulate in the spent fuel. One way to manage these MAs is to transmute them in nuclear reactors, including in light water reactors, fast reactors or accelerator-driven subcritical systems. The transmutation of MAs, however, is not straightforward, as the loading of MAs generally affects physics parameters, such as coolant void, Doppler and burn-up reactivity.

This report focuses on nuclear data requirements for minor actinide management, the review of existing integral data and the determination of required experimental work, the identification of bottlenecks and possible solutions, and the recommendation of an action programme for international co-operation.

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