MURE: MCNP Utilities for Reactor Evolution

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Collaboration : **LPSC Grenoble**

L'équipe MURE de l'IPN travaille sur la simulation de réacteurs nucléaires innovants et sur les scénarios associés. Pour mener à bien ces études, nous avons développé un ensemble d'outil en C++, basé sur le code Monte Carlo MCNO, et appelé MURE (MCNP Utility for Reactor Evolution). Un travail important de développement a été entrepris ces dernières années et a abouti à un ensemble d'outils validés. MURE permet aujourd'hui de simuler de façon complète la neutronique d'un système innovant, de calculer les paramètres de sûreté et l'évolution précise du combustible. Le couplage de MURE avec un code de thermo -hydraulique est en cours, et doit aboutir à la possibilité de mener des études détaillées de la sûreté des systèmes innovants en simulant en 3D des transitoires rapides.

MURE

Numerical studies of a reactor's neutronics can be based on deterministic codes or Monte Carlo codes. The advantage of the former ones is that they are very fast and accurate but only for well known systems such as conventional UOx PWR (Pressuriezed Water Reactor) or U/PU FNR (Fast Neutron Reactor). They solve the diffusion equation by discretization of space and thus time using physical models based on assumptions (that are validated on existing reactors) which lead to simple geometry descriptions (essentially 2D).

The study of innovative reactors required the used of accurate numerical codes, independent of models, using Monte Carlo techniques which depend only on the nuclear data libraries (cross-sections, ...) which allow full 3D description of the exact geometry. Monte-Carlo codes require more CPU time for calculations (but are not prohibitively long) and are the only reference for innovative reactors or fuel cycles. We have developed a number of tools to couple a Monte Carlo transport code for neutronics, MCNP (which is one of the most validated and used codes around the world in this field) with fuel burn-up, called MURE (MCNP Utility for Reactor Evolution) . This work has taken place in collaboration with LPSC Grenoble, Subatech Nantes and NRI Czech republic. MURE is written in an oriented object C++, allowing greater portability and conviviality.

In particular, the MCNP geometry input and temperature dependent cross-sections can be defined very easily. MURE''s aims are neutronics experiment simulations, time evolution of reactor fuels and transient and safety analysis. The main part of MURE concerns the burn-up calculation. It consists of a set of MCNP calculations to obtains fluxes and mean cross-sections for given reactor material composition, followed by a numerical (Runge-Kutta) integration of the Bateman equations governing the reactor fuel evolution (fissile burn-up, appearance of fission products and minor actinides) using MCNP results and providing the new composition for the next MCNP run.

Special attention has been paid to the definition of evolution conditions: power profile over the evolution, reactivity control by either neutron poisons or control rods, refueling capability, ... After the evolution a graphical interface based on ROOT allows a global view of results as well as offers the possibility to make post treatement (radiotoxicity, heat, ...).

In the following section, a description of the first attempt at safety and transient studies is shown, and these have been performed by coupling a the thermal hydraulic code, COBRA-en with the neutronic part of MURE..

The modularity of MURE has allowed us to perform extensive tests and benchmark to validate each part of the code: PWR UOx assembly, Sodium Fast Reactor assembly, Candu evolutions have been studied and show a pretty good agreements leading to a great confidence in MURE. As a proof of the MURE quality, one can mention that it is used in 4 CNRS laboratories, at CEA, at EDF R&D as well as in laboratories of Czech republic, Hungary, India, USA, Canada, Russia, Netherland,

MURE calculations provide an accurate study of innovative reactors as well as realistic input data for global scenarii (waste productions, fissile inventories, ...) of the future of the nuclear power .

It is considered to couple the neutronics results of MURE with a validated scenario code (COSI, developed at CEA), able to perform detailed calculations of the main characteristics of equilibrium or transition scenarios: produced wastes, fissile inventories, reprocessing capacities, impacts on reprocessing and fabrication plants, …

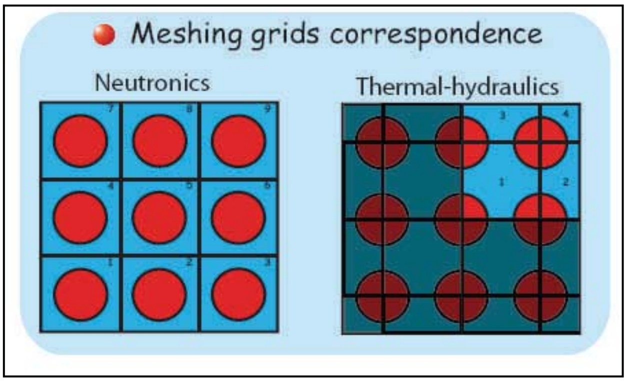
3D coupling neutronics and thermal-hydraulics

A nuclear reactor is mainly simulated by two disciplines: neutronics and thermal-hydraulics. In fact, reactor core temperatures depend on the heat sources, and thus on the distribution of power which evolves over the course of time and is calculated by neutronics codes. Yet, these same codes require cross sections which depend on the temperatures, and thus of the nature of coolant

flow which is solved by thermal-hydraulics codes.
To fully simulate a reactor therefore requires the simultaneous simulation of two very different physics problems: Neutron transport and fluid flow/ heat transfer. For this reason the two codes must be coupled.

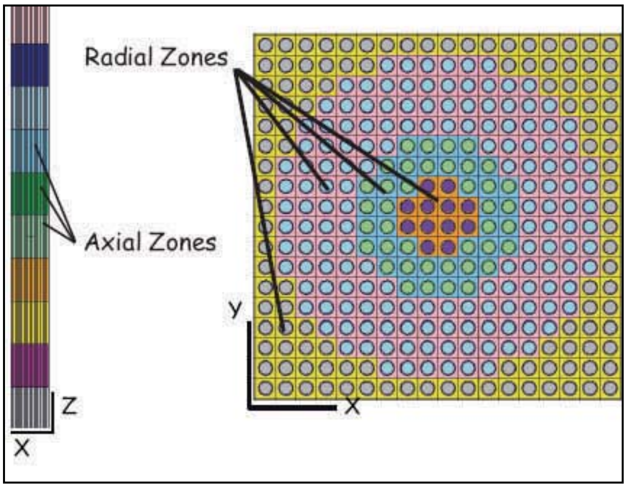
For the neutronics part of the simulation, MURE (MCNP Utility forReactor Evolution) is used to determine neutron flux and reactions rates which vary as a function of space and the time.

For the thermal-hydraulics part, the Oak Ridge National Laboratory code COBRA-EN was selected. It is a sub-channel code that allows steady-state and transient analysis of the coolant in rod arrays. The simulation of flow is based on a three or four partial differential equations : conservation of mass, energy and momentum vector for the water liquid/vapor mixture (optionally a fourth equation can be added which tracks the vapor mass separately). The heat transfer model is featured by a full boiling curve, comprising the basic heat transfer regimes: single phase forced convection, sub-cooled nucleate boiling, saturated nucleate boiling, transition and film boiling. Heat conduction in the fuel and the cladding is calculated using the balance equation.
The input file for a coupled analysis is the same that a neutronics MURE calculation. Some details must be added (like coolant temperature entrance, pressure,...) but a non thermohydraulician can do quickly a simulation with a thermal-hydraulics coupling. The build of the COBRA input file is done automatically and without any intervention by the user, like the coupling procedure between MURE and COBRA.



The neutronics code calculates the power distribution in the fuel pins, which is transferred to the thermal-hydraulics code. Then this last determine the new properties in each pins and sub-channels (temperatures, densities,...). Finally, this data are used as new input for the next neutronics calculation. Iterations can be repeated until a converged state is obtained or during a step time for an transient analysis.

Meshing grids are different between neutronics and thermal-hydraulics. In fact two different physics disciplines are deal with, which solving methods are not linked. It's the reason why we have two different meshes. Consequently subroutines manage the correspondence and the symmetrisation of these meshes. Updates are done using mean values by zones. These zones are chosen by the user and dependent on the precision required. In each radial zone and axial level, neutron flux, reaction rates, properties of materials or coolant and fuel burn-up are treated separately. Any increase in the discretization must be considered carefully because it may use much more computing resources for a marginal accuracy gain, or even, in certain circumstances a loss of precision.



After iterations between neutronics and thermal-hydraulics at steady-state, we obtain calculations results with an acceptable accuracy (cf. fig 3). Further studies will be performed to validate the coupling approach.

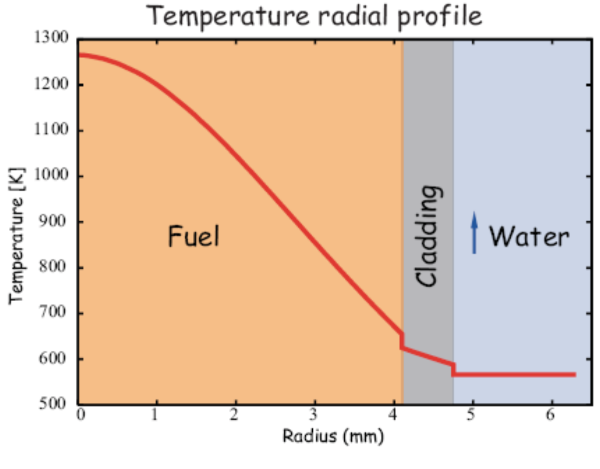


Figure 3: Temperature profile

Our interests are on innovative reactors based on thorium cycle. Light water reactors like advanced Boiling Water Reactor fuelled with Th/Pu for production of ^{233}U are our priority. But Pressurized Water Reactor using any cycles can be simulated, and investigations will be done to allow simulation of Super Critical Water Reactor with Th/Pu fuel. The coupling of MURE/MCNP - COBRA is currently being finalized. The goal of this coupling is a generic, non system specific code, for both burn-up calculations and also safety analysis. These analyses can be carried out at any point in the fuel cycle, because the eventual trajectory of an accident scenario will be sensitive to the initial distribution of fissile material and neutron poisons in the reactor. The addition of point kinetics and the incorporation of delayed neutrons into the calculation will allows simulations of dynamical situations and transients (such as accidents involving : Loss Of Coolant Accident, Transient Over Power, Loss Of Flow, OVer Cooling,...).

After 5 years of development, MURE is today a useful tool which allows detailed neutronics calculation on innovative nuclear reactors. It can also be used for others application, as experiment pre-simulation, thanks to its powerful geometry interface. The development consists now to couple the code with a thermal-hydraulics code, in order to perform detailed studies of safety. Preliminary results have been obtained and we are validating a first coupling with COBRA.

Thanks to MURE, the IPN team performs different studies of innovative systems. On one hand, we focus today on high conversion ratio reactors, based on the well-known water technology, and using a Th/Pu or Th/^{233}U cycle. On the other hand, we perform detailed calculation on sodium fast reactors, based on Uranium cycle or Thorium cycle. The results of these neutronics studies must be provided to COSI (CEA) in order to perform detailed calculations on the whole scenario (transition and equilibrium) and to determine optimized strategies minimizing global waste production and fissile inventories.

MURE Simulations: The Thorium Cycle. Fast Reactors

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Collaboration : **LPSC Grenoble**

L'ensemble d'outils MURE, développé a Orsay en collaboration avec le LPSC Grenoble a été utilise pour deux études. La première étude concerne la simulation d'assemblages et l'application à un scénario à trois composantes utilisant les cycles Uranium/Plutonium et Thorium/Uranium. Le parc n'est pas régénérateur, mais permet d'économiser considérablement les ressources en uranium et de limiter la production de déchet, ceci en utilisant la technologie actuelle basée sur l'eau légère ou l'eau lourde.

La deuxième étude concerne la simulation d'un réacteur à neutrons rapides refroidi au sodium, qui permet de réduire d'un facteur 100 la consommation de minerai d'uranium et de mettre en œuvre la transmutation des actinides mineurs. Les simulations en cours vont permettre de valider MURE pour les réacteurs rapides et doivent ensuite être couplées à COSI (code de scénario du CEA) afin de mener des études de scénarios détaillées.

Currently there is a renewed interest in the expansion of the role of nuclear power for energy generation due to its lack of CO_2 emissions (which contribute to global warming), its economic viability, and its potential for energy security and independence. However, as developing countries rapidly increase their installed nuclear capacity and easily extractable Uranium resources become depleted, the result will likely be much higher Uranium prices over the course of this century and a legacy of large quantities of unburied nuclear waste.

We consider two types of strategies: the well-known water technology and the use of fast breeder reactors. In both cases, the advantages of the thorium cycle are explored.

Economy of Uranium resources in a three-component reactor fleet with mixed Thorium/ Uranium fuel cycles

Over the last 50 years the pressurized water reactor (PWR) has become ubiquitous due to its safe and reliable operating characteristics derived from its water coolant/moderator, a well-understood, non-corrosive liquid with a high heat capacity.

Unfortunately, the simple scaling up of current PWR technology is not a sustainable strategy for nuclear power development and expansion in the long term and will furthermore massively compound the problem of nuclear waste disposal. Currently there is a renewed interest in the expansion of the role of nuclear power for energy generation due to its lack of CO_2 emissions (which contribute to global warming), its economic viability, and its potential for energy security and independence. However, as developing countries rapidly increase their installed nuclear capacity and easily extractable Uranium resources become depleted, the result will likely be much higher Uranium prices over the course of this century and a legacy of large quantities of unburied nuclear waste.

While widespread use of fast breeder, closed-cycle reactor technology remains an ideal long-term solution to the Uranium resource and radioactive waste disposal problems, this century it is highly unlikely that the global nuclear fleet will expand its collective breeding ratio to greater than unity from a current value of ~ 0.4, due to the huge capital costs and technological difficulties involved in such fast breeder systems (e.g. corrosive liquid metal coolants, and positive void coefficients) and the long time-constants involved in their development and large-scale deployment. We show that a fleet of reactors using systems of *three* components involving mixed Uranium/Plutonium and Thorium/Uranium offers great scope for the both economizing on the use of Uranium resources (up to 80% reduction) and producing significantly lower quantities of minor actinide waste while relying on only inexpensive water-based reactor technology.

We envisage an intermediate stage in the global development of nuclear power where reactor fleets consisting of three different components optimize the neutron potential of water-based reactor technology and thus minimize Uranium resource consumption. While such three-component systems will have breeding ratios of less than unity, they are likely to be more economically competitive than a fleet of either conventional PWR's *or* a fleet of fast breeder reactors over a range of Uranium prices between roughly estimated to be between \$300/kg and \$1000/kg.

The introduction of the Thorium cycle offers several advantages, principally among them the production of up to 1000 times less minor actinide waste in a thermal spectrum reactor and the possibility of high breeding ratios due to the low alpha (capture/fission) ratio of the fissile ^{233}U. However, a major drawback is that since ^{233}U does not exist in significant quantities in nature it must be first produced and its quantity sustained by running Th/^{233}U breeder reactors (breeding ratios BR ≥ 1.0). However, since the neutron balance is so delicate in such systems, breeding can only be achieved through radically innovative designs: either molten salt reactors (e.g. MSBR[1]) with complicated on-line chemical reprocessing to remove neutron poisons or reactors similar to the Shippingport[2,3] design with geometrically inconvenient and awkward heterogeneous cores. Our solution to this problem is to dispense with the need to develop technologically innovative and economically unviable Thorium breeder reactors and instead run a reactor with a more conventional design with Th/^{233}U fuel in *near* breeder mode (BR < 1.0). Achieving such breeding ratios is a perfectly feasible goal using standard water-based technologies. However, this obviously necessitates the production of ^{233}U elsewhere in the reactor fleet to make up for the deficit of ^{233}U incurred at each fuel reprocessing.

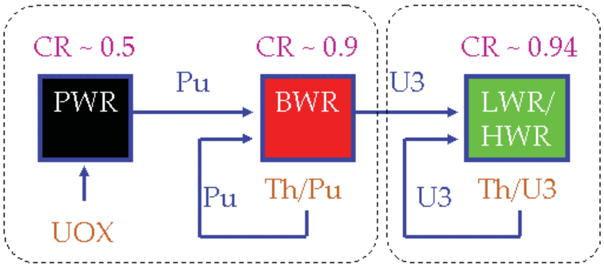


Fig. 1. The three component reactor fleet with approximate conversion ratios at each stage, and the option of multi-recylcing the Plutonium at the second stage.

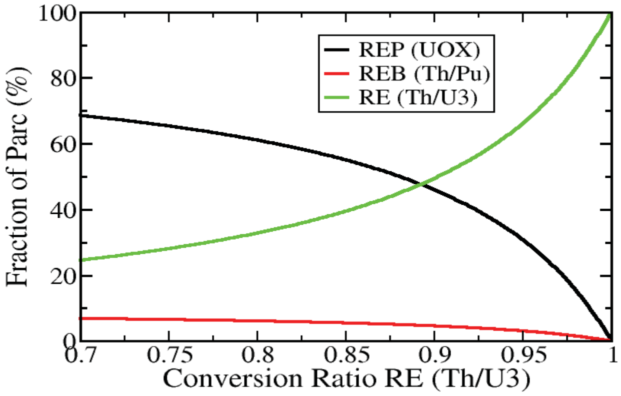


Fig. . The depedency of the fraction of the reactor fleet that consists of the final U3 consumer stage on the conversion ratio of the third type of reactor. The utilisation of Uranium resources is directly proportional to the black curve.

The first part of the three component is the conventional PWR's (which currently exist) loaded with standard Uranium Oxide (UOX) fuel and reprocessed at the end of the fuel cycle to extract the Plutonium and fabricate Thoriated fuel for the second component, a Pu/^{233}U converter. To maximize production of ^{233}U at this stage an epi-thermal neutron spectrum is needed to maximize neutron capture reaction rates on Thorium, so the second component is chosen as either a boiling water reactor[4] (BWR) or super-critical water reactor[5] (SCWR) loaded with Th/Pu mixed oxide (MOX) fuel. The ^{233}U extracted from the converters' spent fuel is then fed to the third component, a light or heavy water reactor (LWR or HWR) designed with a high breeding ratio (BR~ 0.9 – 0.95 dependent on the fuel cycle duration) and loaded with Th/^{233}U fuel which is then multi-recycled (see figs 1. & 2.)

Modeling of the burn-up of fuel in the assemblies of each of the reactor components in such three-component systems has been carried out using the MURE[6] Monte-Carlo precision evolution code based around MCNP[7] and developed at IPN Orsay and LPSC Grenoble. We show that for reasonable fuel cycle lengths and reprocessing assumptions it is possible to reduce the consumption of natural Uranium by the reactor fleet at equilibrium from 200 tons/y/GW of installed capacity (consumption for a standard PWR fleet running in open-cycle mode) to values as low as 40 tons/y/GW for the 3-component system. However, figure four shows that the economically, longer burnups and lower conversion ratios are favoured.

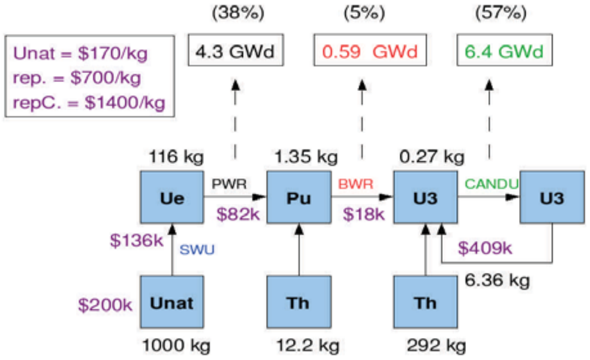


Fig. 3 Typical mass flow diagram for the 3-component reactor fleet, along with approximate economic costs at each stage. The final stage acts as an energy amplifier with a gain of 1/(1-CR).

Such three component systems could be developed rapidly due to the fact that the PWR's and in some cases BWR's already exist in the current fleet's of reactors. The proposed technology, although requiring some innovations in fuel reprocessingnd reactor safety, could be deployed within relatively short time-scales of 20-30 years. It would also not hinder the future development/deployment of fast breeder reactors if the need for such a transition were to arrive. In addition, if collaboration between countries with civil nuclear power programs was increased and restrictions on the transport of fissile materials between countries lifted, then a prototype 3-component system may already exist in Asia: Spent fuel from China's PWR's could be used to produce ^{233}U in Japan's BWR's, the reprocessed fuel being used to provide Th/ ^{233}U for multi-recycling in South Korea's CANDU reactors. At no stage in the fuel cycle of the three-component system will fuel be fabricated with a fissile content of greater than 6%, so the proposed development is relatively proliferation resistant and is consistent with the aims of the Global Nuclear Energy Partnership (GNEP)[.

To conclude, we suggest that the potential for the expansion of the role of nuclear power by development of three-component reactor fleets and mixed Thorium/Uranium fuel cycles based on improvements of existing water-based technology is both feasible and realistic. The deployment of such systems is not dependent on huge leaps in technological progress and/or the successful construction/demonstration of radically innovative designs and could lead to a significant economy of Uranium resources and large reduction in volumes of radioactive waste produced.

Simulation of Sodium Fast Reactors and innovative associated scenarios

Fast reactors are an interesting option for the future of civilian nuclear power. They allow the implementation of breeding in order to decrease the natural uranium consumption by over a factor of 100 as compared to current technologies. They also allow the possibility of the transmutation of minor actinides which can significantly reduce the inventory of long-lived radioactive wastes.

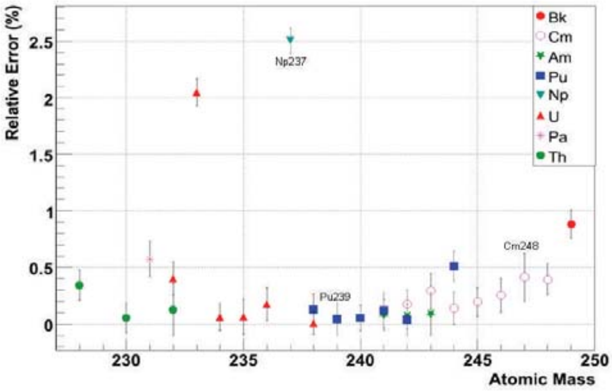
The work performed consists on simulating different sodium fast reactors (SFR) with the MURE code. The first study deals with standard concept of breeding sodium fast reactor in order to validate our simulation method. The results concern system safety coefficients calculating, key parameters, that still stay to improve, and fuel evolution in core and regenerated rate evolution.

A study on setup calculations has also been performed in order to find the best compromise between calculation time and precision of the results. Several tests may be implemented such as:

- convergence of neutrons source ;
- number of evolving nuclei ;
- number of MCNP calculation;
- material homogenization.

Moreover, a study on error propagation has been performed after 5 years evolution on a fast reactor U/Pu assembly. We have made the following observations:

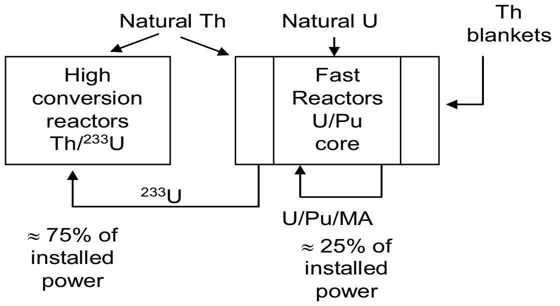
- for nuclei of interest, the error is lower than 0.5%;
- the further we move away from nuclei initially present, the more the cumulated error increases (Cf, Bk, ...)
- for nuclei coming from (n,2n) reactions, that are reactions at a threshold, relative error is larger (because we are in the spectrum zone where there are few neutrons).



Finally, a benchmark with SFR is currently being performed. A core calculation and safety parameters calculations will use an exact 3D exact geometry.

The perspective of this work is to explore the potential of sodium fast reactors for transmutation of plutonium and/or minor actinides, and for ^{233}U production in thorium blankets. This option could provide a way to minimize fissile inventories and waste production by using a symbiotic park of two components, as shown on the following figure.

The MURE results must be used to simulate this kind of equilibrium scenario with COSI (CEA), as well as the transition phase starting from present situation : calculation of accumulated wastes, inventories of fissile materials, annual material fluxes in reprocessing plant, .



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Fission cross sections of actinides at nTOF, from resonances to spallation

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Mesures de sections efficaces de fission à nTOF, des résonances à la spallation
Le faisceau de neutrons de l'installation nTOF du CERN offre des possibilités uniques : gamme en énergie étendue (0.1eV à 1GeV), très bonne résolution en énergie des neutrons, haut flux instantané particulièrement utile pour les cibles radioactives. Nous avons mesuré les sections efficaces de fission par détection en coincidence des fragments de fission, ce qui permet d'éliminer la contribution des autres types de réaction à haute énergie, et réduit le bruit de fond de radioactivité. Les isotopes pour lesquelles les mesures ont été effectuées sont ^{232}*Th,* ^{233}*U et* ^{234}*U jouant un rôle important dans le cycle du thorium,* ^{237}*Np comme cible possible de l'incinération, ainsi que* ^{nat}*Pb et* ^{209}*Bi qui interviennent dans les cibles de spallation. Toutes les mesures ont été effectuées relativement à* ^{235}*U et* ^{238}*U considérés comme références.*

The development as well as the design of new nuclear reactors require a more precise knowledge of nuclear data than presently available. The nTOF facility at CERN allows precise measurements of neutron-induced reactions due to its long flight path (185 m), its very broad neutron energy range (from 0.5 eV to 1 GeV), and its high neutron flux delivered in a very short time, particularly interesting for the study of radioactive isotopes.

Measurements have been performed first for isotopes which are of interest for the thorium cycle: ^{232}Th, ^{233}U and ^{234}U. Concerning minor actinides ^{237}Np has been measured because it is a possible target for incineration. In addition ^{nat}Pb and ^{209}Bi have also been measured as preferred materials for spallation targets. All measurements are performed in reference to ^{235}U and ^{238}U permanently present in the stack of targets.

The detection system was built in order to detect the 2 fragments in coincidence and hence discriminate fission events from other types of reaction : radioactivity, recoils of reactions appearing at energies above 10 MeV, and spallation reactions at a few tens of MeV. The main constraints induced are the need of very thin backing for the targets and very thin electrodes for the detectors. Due to the limitation of the angular acceptance of fission fragments by their stopping in backings and electrodes we had to control this acceptance by reconstructing their trajectory [1].

We used PPACs (Parallel Plate Avalanche Counters) because they can be designed with thin electrodes. In addition their good timing properties are valuable for coincidences and they can deliver a XY localisation information allowing the trajectory reconstruction [2] requested for the angular acceptance monitoring and also useful to correct for spatial variations of the neutron flux and inhomogeneity of the targets. From the localisation points on each detector the full trajectory of the event is reconstructed, assuming a back to back emission of the fragments.

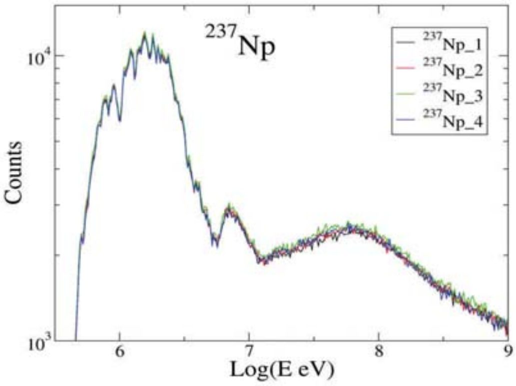

Fig 1 : Counting rate obtained using the anode signals only for 4 ^{237}*Np targets as a function of the logarithm of the neutron incident energy. Data are corrected for the different targets thickness.*

The experimental setup is made of a stack of 10 detectors interleaved with 9 targets including the 2 reference isotopes. The ambiguities arising from the possible crossing of 2 detectors by the same fission fragments are solved by the coincidence time between detectors.

The detailed characteristics of the targets are described in [2]. The a activity of some of the targets allowed a measurement of their homogeneity using a counting ; for the less active materials, the RBS technique was used. The consistency between the two methods was successfully checked in the case of the ^{235}U target.

Due to pile-up problems, localisation measurement could not be used for a systematic monitoring of the detector acceptance. Therefore the cross sections measurement rely only on the coincidences of the fast anode signals. However, as it can be seen in the figure 1, the counting rates

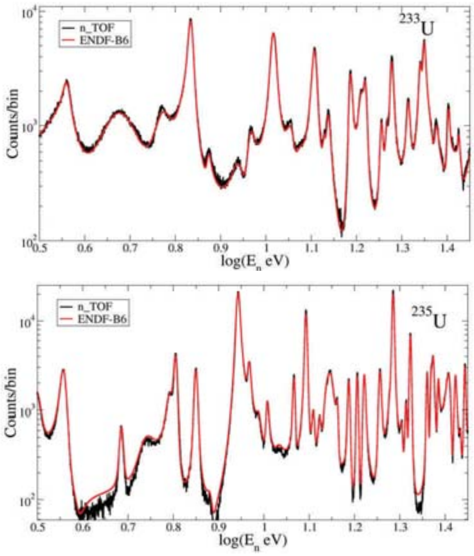

Fig. 2. Fission counting rate measured at n TOF (continuous line) as a function of the neutron energy, compared to ENDF-B6 values (dashed line) - see text for details.
Top figure : ^{233}U. Lower figure : ^{235}U.

obtained with this method for 4 different ^{237}Np targets are in good agreement. The inconsistencies between these rates indeed provide an excellent estimation of the systematic error on the counting rates. We evaluate this error to 5% below the MeV region, 10% in the tens of MeV region (this is a conservative value aimed at accounting for the possible different variations of the fission anisotropy from one nuclei to the other), and 7% beyond 100 MeV (again a conservative value taking into account the efficiency variations caused by the recoil imparted to the fissioning nucleus by the momentum transfer).

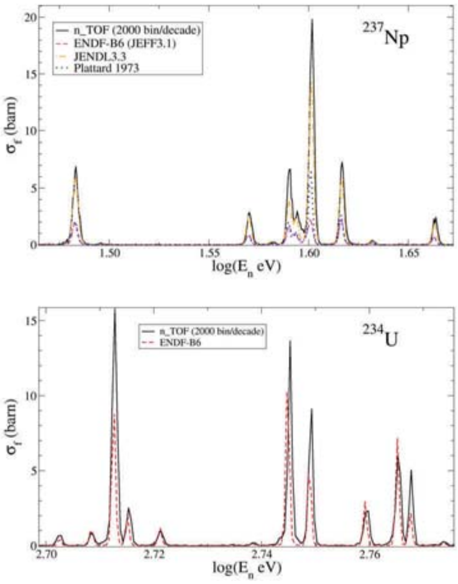

Fig. 3. Lower figure : fission cross section measured at n TOF on ^{234}U (continuous line) compared to ENDF-B6 values (dashed line) as a function of the logarithm of the neutron energy.
Upper figure : same for ^{237}Np, with additional plotting of JENDL-3.3 (dashed-dotted line)
and Plattar measurement (dotted line).

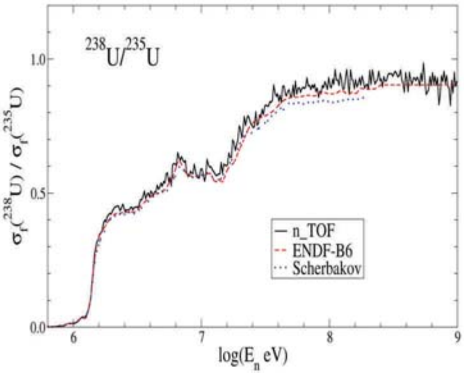

Fig. 3. Lower figure : fission cross section measured at n TOF on ^{234}U (continuous line) compared to ENDF-B6 values (dashed line) as a function of the logarithm of the neutron energy.
Upper figure : same for ^{237}Np, with additional plotting of JENDL-3.3 (dashed-dotted line)
and Plattar measurement (dotted line).

The high energy resolution and the low neutron background of the nTOF facility are revealed in figure 2 which shows the fission rate in the energy interval between 3 eV and 30 eV for ^{233}U and ^{235}U. These results indicate that the n TOF measurement has been carried out with better neutron background conditions and statistics than previous measurements. At higher energies, the improvement of the resolution allows a new evaluation of the fission cross sections in the resonance region [3].

In the case of ^{237}Np and ^{234}U, there are clear discrepancies between our data and evaluated databases in the subthreshold resonances region (see figure 3). Some of these resonances from ^{237}Np appear to be underestimated by factors between 5 and 10 in ENDF-B6, and by 25% in JENDL-3.3 (upper figure). Discrepancies are more reduced, while still significant, in the case of ^{234}U resonances.

At higher energy, the ratio of fission cross sections of ^{238}U and ^{235}U is an interesting test case, as in our case, this quantity is independent of the neutron flux because the measurements on both nuclei were performed simultaneously. Our results are presented in figure 7. We find that our data are very close to the ENDF-B6 + JENDL-HE base (note for example the nice agreement in the delicate shape associated with the 2nd chance opening), with a systematic shift of about +5%. Although this shift is smaller than our estimated systematic uncertainty, it is worth noticing that the preliminary results obtained by FIC [4], also at nTOF, show the same tendency. If we compare our results to previous measurements, we find the same slight shift with the work of Lisowski (on which the evaluations largely rely), and Shcherbakov [4] beyond 20 MeV. It is difficult to draw solid conclusions from such discrepancies as all these results are compatible within their respective error bars.

Future measurements will focus on more radioactive targets such as Am isotopes. We also plan to dedicate measurements to the fission anisotropy phenomenon, which will be of interest for the theoretical understanding of the fission and will also help to reduce the uncertainties on our data in the 10 to 100 MeV region. Absolute fission measurements would be highly relevant, especially for the high energy part ; but such a measurement requires an appropriate neutron detector, probably based on the (n,p) diffusion for which the cross sections are very well known. We have started to explore the feasability of such a detector for the n TOF facility.

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CACAO: a laboratory project for the production and characterization of radioactive targets

IPNO Participation: **Ch.O.Bacri, V.Petitbon, S.Pierre, J.F.LeDu, Y.Adès, N.Vigot, et. al**

L'objet du projet CACAO (Chimie des Actinides et Cibles radioActives à Orsay) est la construction d'un laboratoire national de production et de caractérisation de couches minces radioactives. Ce projet est né des besoins des physiciens nucléaires issus de la communauté travaillant sur le cycle électronucléaire (aval du cycle et réacteurs du futur), mais aussi de ceux des physiciens provenant de la communauté d'astrophysique, et de ceux qui travaillent sur la synthèse et l'étude des éléments superlourds. Les caractéristiques des cibles demandées ainsi que la nécessaire caractérisation de ces cibles ont permis de dimensionner les installations nécessaires ainsi que leurs futures fonctionnalités. Outre les contraintes de sureté et de radioprotection, la principale difficulté provient de la perte du savoir-faire nécessaire à la réalisation de telles cibles. Nous sommes actuellement en train de définir de façon précise les conséquences d'un tel laboratoire en terme de besoins en personnel et de financement ainsi que de définir les différents partenariats. entre laboratoires du CNRS et du CEA qui seront impliqués

Introduction

The CACAO project, Chimie des Actinides et Cibles radioActives à Orsay, is still under construction. It consists on the installation of a hot laboratory dedicated to the production and the characterization of thin radioactive layers. The starting point of the project was the experimental program developed at IPNO, in the PACS group, which is induced neutron fission studies, related to nuclear waste management, and new fuel cycle. This program is developed on the nTOF line, at CERN. In this frame, we are engaged to measure fission cross section for different isotopes: ^{233,234,235,238}U, ^{232}Th, ^{237}Np and ^{241,243}Am.

For this purpose, the needed targets induce some constraints on their characteristics: very thin targets with very thin backing, in order to detect both fission fragments, and very large targets, in order to optimize the use of the 8 cm diameter neutron beam of the nTOF line. At last, high purity samples are needed, especially because any contamination with fissile element is of course forbidden. These constraints lead to such difficulties that we finally produce ourselves some of the targets (^{233,234,235,238}U, ^{232}Th, ^{237}Np), thanks to our radiochemistry group (see fig.1).

Indeed difficulties related to the production of radioactive targets are very common, and a potential user is usually confronted to some difficulties which can be summarized as follow:

- the loss of knowledge for the production of thin layers on thin backing
- the specificity of the different laboratories which are able to produce these targets. This leads to many difficulties for the choice of the good lab, especially when you do not understand all the techniques and their respective advantages.
- very often, it is difficult to fulfill recurring needs, because usually laboratories are dimensioned with respect to their own needs.
- it is difficult to obtain precise characterization of the target. The best way to know precisely the homogeneity and the chemical and isotopic contaminant of the target is to measure it yourself. Safety problems and the related lack of installations lead those measurements very difficult.
- transport of radioactive targets is not always easy, especially when you have to cross a frontier

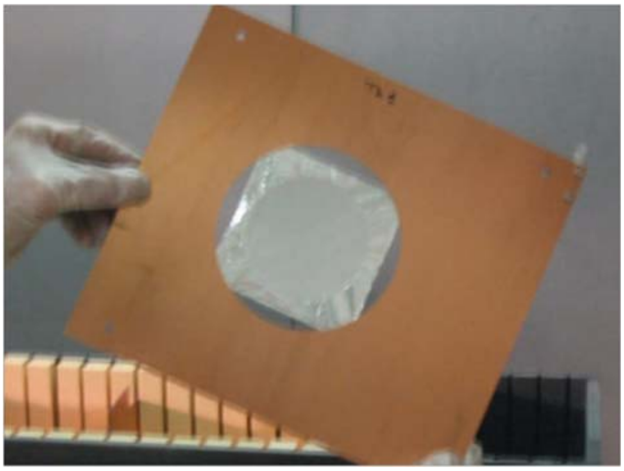

Fig.1: ^{235}U target, made of a deposit of 8 cm diameter of .3 mg/cm^2 onto an Aluminum backing of 2 mm.

Finally, the most important difficulty is the lack of coordination between all the different steps of the production of the target.

In order to work out this complexity, a project of hot laboratory of production and characterization of radioactive target is under construction at Orsay.

We have indeed at Orsay a radiochemistry group specialized in actinide chemistry, and, even if it develops its own research program, it has the needed knowledge to produce targets. Moreover, we have also a target laboratory which produces stable isotopes thin layers, and this knowledge is indispensable when thin backing are needed for the targets.

The first task of the lab will be to help the users to supply isotopes, and if needed, an isotope separation will be possible. Then, we will produce targets, either by radiochemistry techniques, either by direct implantation, and then, it will be possible to characterize these targets, to make a cartography of it, to determine different contaminants, and so on. At last, we will help for the organization of the transport to the experimental installation.

As this kind of installation is difficult to setup, essentially for safety reasons, but also because it will be expensive, the aim of the project is to setup a joint CNRS-CEA national laboratory. Of course, it will be important to be complementary to already international installations.

In order to collect all the needs of the community and to be sure that the needed investment will be justified, an inquiry was done, mainly towards the French community. This inquiry had also helped us to the precise outlines of the future installations in a way which answer to the expressed needs. The main results of this inquiry can be summarized as follows:

- Up to 52 isotopes were asked, from ^{14}C to ^{252}Cf. This very large variety leads us to not restrict the laboratory to actinides only, which are our own needs in my group. It reflects the large community interested by this laboratory, going from astrophysics, to nuclear data for fuel cycle, through super-heavy element community.
- Some demands are recurrent, up to 5 targets per year during 10 years. They mainly concern transuranium elements.
- The wanted thicknesses go from some hundred mg/cm^2, up to some hundred mg/cm^2.
- There is clear requirement for isotopic and chemical purity, or at least for a good knowledge of them.

Different isotopes suppliers were already identified: industry and some research laboratories. It will be also probably possible to produce isotopes by irradiation inside the OSIRIS reactor, at Saclay. Moreover, the possibility to use an accelerator to produce isotopes is under study.

The isotopic separation is a more difficult point to solve. Indeed, for safety problems, there is no available magnetic separator for radioactive isotopes in France, allowed to handle with “macroscopic quantities” of radioactive isotopes.

Different solutions are the under study, including the possibility to construct a new dedicated separator, and we foresee to take a decision at the beginning of 2009.

Considering the production of the targets, the main constraint which will size the installation is that it is important to be able to use the targets in a beam line. This leads to restrict installations to glove boxes in controlled zone. We have obtain from the EFNUDAT European I3 (see http://www.efnudat.eu) 12 months of fixed term contract in order to transfer the know-how of our radiochemistry group. The goal is then to transform this contract into a permanent position dedicated to the new hot lab. In parallel, we have also collaboration with the Berkeley University in order to complete our knowledge in manufacturing of targets.

At last, the foreseen installations will allow for the target characterization, that is cartography and homogeneity measurements.

More technical information can be found in the technical part of this report, under the article of Véronique Petitbon.

The project was approved by the Scientific Council of the Nuclear Department of CNRS. In order to be able to really begin the realization of the project, we are now working to produce a complete document in which will be detailed the consequences of the project in terms of budget and manpower, and in which we will identify the partners of the project. We plan to realize the first targets for the community on the horizon 2010.

Energy and environment : a civic course of physics

IPNO Participation: S. Bouneau, S. David

L'objectif majeur de ce cours est de fournir aux étudiants qui se destinent à poursuivre un parcours scientifique à l'université des bases solides pour mieux appréhender la crise énergétique que connaît actuellement le monde, en termes de production et de consommation d'énergie, et son impact sur le dérèglement climatique. Le cours propose des calculs concrets des besoins énergétiques (chauffage d'une habitation, transport, électroménager, ...) et des performances des différentes sources d'énergie capables de les satisfaire (panneaux solaires thermiques, combustibles fossiles, ...). Ce cours vise également à proposer une façon concrète d'aborder de nombreux thèmes de physiques afin de rendre plus attractive cette discipline de plus en plus délaissée par les étudiants.

Since the start of the academic year 2006, a new optional course (25 h) is proposed to scientific students at different level from the first year (L1) up to the third (L3). The main aims of this course are the following :

- make scientific students aware of the energy challenge that the planet must face with today in terms of production and consumption, and their impact on the environment;
- give to students some tools to apprehend this challenge in a scientific way;
- teach physics in a practical approach by applying concepts and theoretical laws to real cases of everyday life.

The energy problem is complex and difficult to set down in a proper way. To prove, the media give very frequently wrong and confused information on the subject through lack of scientific basis. Thus, one of the motivations of this course is to give to the students the fundamental basis of physics to understand what we call the "world energy challenge" and to be able to make simple calculations to have some orders of magnitude in mind. Firstly, an important part of the course is devoted to describe the present and future energy landscape. For example, actually 80 % of our energy consumption comes from fossil fuels (coal, gas and petrol) responsible of massive emissions of carbon dioxide (CO_2) which represent the main contribution to the climate change. The emerging countries (China, India) are in full economical expansion and the part of the population claiming rightfully the same life level than the one of actual developed countries will increase significantly. How to face with an explosive energy demand when the middle of the century will probably see the decrease of petrol and gas production ? What are the alternatives to replace the use of fossil fuels with "clean" and renewable energy sources to minimize the human activities impact on climate change ? To attempt to answer to these questions from a physics point of view, we stress first of all on the fundamental law to which energy and all its transformations or transfers obey : the energy conservation. This concept introduces the fact that we cannot create energy, all we can do is to transform the energy initially available (sun, wood, wind, petrol) in other forms that we can use for our needs (heating, mechanical energy for transport, electricity). Whatever the machine or the installation used to achieve these energy transformations, the physics law impose that a certain amount of energy cannot be used and must be considered as lost. This waste of energy inherent to its transformation is quantified by the efficiency of the machine which cannot be of 100 %. In the course, an overview of all energy sources available is made and their transformation in useful form for our needs is described in details. The approach systematically adopted is the following :

Identification and quantification of our needs in energy. For example: complete calculation of house heating including the different transfer modes of heat and calculation of energy consumption for transport.

- Available energy sources : Identification of the energy form of the source. For example : chemical (fossil fuels, biomass), light (solar), potential (hydraulic), kinematical (wind).
- Examples of heat machines to transfer or transform the heat into mechanical work or electricity : refrigerator, heat pump, motor, power plant
- Examples of energy sources and their use for our needs : basis of nuclear energy for electricity generation (fission, reactor principle, nuclear waste) ; solar energy and complete calculation of the efficiency of heating panels working at low temperature ; fossil fuels for heating and transport, and calculation of the corresponding CO_2 emissions.

This field allows us to cover several physics concepts that students are led to encounter again along their degree course. We hope that this practical approach makes the physics more attractive and that students have a scientific basis to build their own vision on the world energy challenge.