The place of EOLE, MINERVE and MASURCA facilities in the R&D Activities of the CEA.

P. FOUGERAS[1] – A. CHABRE[2]– C. MERGUI[1]

[1] *CEA Cadarache, DEN/DER/SPEx, 13108 Saint Paul lez Durance, France*
[2] *CEA Saclay, DEN/DSOE, 91191 GIF sur Yvette Cedex, France*

KEYWORDS : EOLE, MASURCA, MINERVE, ADS, ALWR, ABWR, Integral experiments.

ABSTRACT

The CEA (Commissariat à l'Energie Atomique) is involved in a research program concerning the future of plutonium, waste management and innovative systems exploration. The critical facilities of the CEA Cadarache: EOLE, MINERVE and MASURCA play an important role in the validation of neutronic calculation tools (codes and nuclear data). Most recent programs notably contributed :

- Obtaining a very large and accurate experimental database for nuclides arising in plutonium and waste management (heavy nuclides and long lived fission products).
- Exploring long-lived nuclides transmutation.
- Exploring innovative systems and new concepts in terms of new materials (ABWR, JHR, ..).
- Improving the physics of hybrid systems, involving a sub-critical reactor coupled with an external accelerator (ADS).
- Reducing the uncertainties associated to the prediction of most of the current and new core concept parameters such as GEN IV (Gaz Cooled Fast Reactor : RCG-R).

All theses programs are carried out within the frame of international collaboration.

1. INTRODUCTION

The CEA is deeply involved in a research program concerning the future of plutonium and the waste management. In this framework, specific neutron integral experiments have been defined in the critical facilities of the CEA Cadarache. The programs carried on the last ten years aimed to :

- EOLE facility, qualifying the current calculation scheme tools used in design and safety calculations for Advanced Light Water Reactors loaded with 100% MOX fuel with the MISTRAL program for PWR and the BASALA and FUBILA programs for BWR.
- MINERVE facility, improving the knowledge of the integral absorption cross sections of heavy isotopes and actinides (OSMOSE program), of new absorbers (OCEAN program) and also for the High Burn Up fuel (HTC program) in thermal, epithermal and fast spectra. These studies are used to extend the validity range of the criticality and radioactivity calculational tools (BURN UP CREDIT program).
- MASURCA facility, improving the physical understanding of the physics of hybrid systems in a fast spectrum and of subcritical multiplying media in presence of a well-known external source. The experimental data are used for the qualification of the calculation schemes dedicated to the core configurations designs for Accelerator Driven Systems (MUSE program) and also, in the future, a new core concept of RCG-R (ENIGMA).

2. THE EOLE FACILITY

EOLE is a pool type reactor composed of a cylindrical vessel of AG3 (diameter = 2.3m and Height= 3m) with an over structure of stainless steel able to contain various types of core and related structures. The water circuit has been designed to control the volume, the composition and the temperature (5°C to 85°C) of the moderator.

Recycling 100% MOX fuel in PWRs seems to be feasible if the Hydrogen-to-Heavy-Metal ratio (H/HM) is increased from H/HM~4 to 5 or 6, but sensitivity studies show that the uncertainties on calculated neutronic parameters remain sufficiently high to exceed some safety requirements. Moreover, new concepts of high moderation 100% MOX cores require an exhaustive experimental program to verify and improve the current calculational tools used for design and safety.

That is why the MISTRAL program **/1/**, **/2/**, **/3/** has been undertaken in the EOLE facility in order to measure the main neutronic parameters of Advanced Light Water Reactors with high moderation 100% MOX fuel cores as a collaboration between NUPEC

(Nuclear Power Engineering Corporation - Japan), CEA and their associated industrial partners. This program consisted in four specific cores: three regular lattices (one UOX lattice and two MOX lattices) with a moderation ratio (H/HM) varying from 5 to 6, and a mock-up lattice (H/HM~6) simulating advanced 17× 17-PWR assemblies (full MOX) were investigated.

New experimental techniques (modified conversion ratio) and new devices (thermo-regulation system) were developed and implemented for this program, allowing to obtain very precise experimental results reaching the main target defined at the beginning of the CEA/NUPEC collaboration. Most of the experimental results have been checked by the current calculation routes leading to obtain valuable information about the validity of the codes treating the physical phenomena occurring in 100%-MOX ALWRs. Further calculations are currently in progress in CEA to evaluate the ability of these codes to reproduce the experimental results obtained in the mock-up cores with and without absorber clusters.

To complete the MISTRAL program, an experimental program called BASALA **/4/, /5/, /6/** started in July 2000 and ended in August 2002. The goal was to measure the main design neutronic parameters of high moderated 100% MOX 9X9 ABWR (Advanced Boiling Water Reactor) cores and to provide accurate experimental results, obtained in heterogeneous media, to support the validation of methods used for design calculations. The BASALA experimental program consisted in two experimental cores based on lattices simulating hot (H) and cold (C) ABWR conditions.

Poison rod effects, void effects, over-moderation effects, axial and radial power distributions, temperature effects and integral boron worth were studied in this program.

- BASALA-H (2001): The lattice comprised 4 ABWR sub-assemblies of 9×9 PWR MOX fuel pins (1.13 cm pitch) and a 1.24 cm water gap placed between sub-assemblies.
- BASALA-C (2002): The lattice comprised 4 ABWR sub-assemblies of 9×9 PWR MOX fuel pins (1.35 cm pitch) and a 1.40 cm water gap between sub-assemblies. A specific hole was manufactured in the grids in order to study the control blade effects (natural B_4C and metallic Hf).

Criticality and radial fission rate distributions were studied. Thus more than 1200 MOX pins were measured by integral □-spectroscopy allowing to obtain a very large data base with a very good accuracy : the fission rates were measured within ± 1.5% and ±1.1% (1 s.d.) in BASALA-H and BASALA-C cores respectively. Safety parameters were also investigated in BASALA-C in order to meet the reactivity margin requirements : the integral boron efficiency, the isothermal temperature coefficient (ITC) and ABWR-type control blade effects ((Hf and nat B_4C) were obtained with a rather small uncertainty.

The calculations were performed by Monte-Carlo and deterministic codes of NUPEC and CEA **/7/.** The results indicated a rather satisfactory calculation of the criticality : a slight over-prediction was found certainly coming from nuclear data ; the TRIPOLI-4 and MVP codes (MC Codes) showed an error of +480 ±45 pcm and +840±34 pcm on an average, the deterministic codes CITATION and TWOTRAN gave slightly worse results. The radial rate distributions were very well reproduced by the Monte-Carlo codes but "high-efficient" heterogeneities led to underestimate (down to ~ -7%) the power in the fuel pin neighboring the heterogeneity. The positive ITC below 30°C and the integral boron efficiency are rather well reproduced by MVP. Further investigations are currently in progress in CEA with deterministic codes in order to evaluate their capability to design full-MOX BWRs.

BASALA is being completed with an experimental program, FUBILA, planned from January 2005 to march 2006. This program is very important step for the licensing of the 100% MOX BWR OHMA Japan core. FUBILA consists in the simulation of different core configurations (4 heterogeneous sub-assemblies of 9×9 ABWR MOX fuel pins with a concentration of Plutonium varying from 3% to 11.5%). Six core configurations are studied covering a large range of operating conditions of high burn-up MOX BWR cores:

- Reference – 0% void
- 40 % void
- 70 % Void
- Axial Void
- Control blade
- Prospective assembly studies 10×10 (40 % void)

440 Specific MOX pins have been manufactured to complete the FUBILA program: four kinds of MOX fuel pins are used for simulating the high burn-up BWR assemblies (Pu enrichment 3.0wt%, 5.0wt%,8.5wt% and 11.5wt%). The Pu vector must contain 60~70 wt% in [^{239}Pu+^{241}Pu]/[total Pu] coming from LWR spent UO_2 fuel reprocessing.

Two new programs are planned on the EOLE facility (the PERLE and FLUOLE programs) in order to support the studies of standard reflector, heavy reflector, thermal shield for the PWR (1300MWe) and the future EPR.

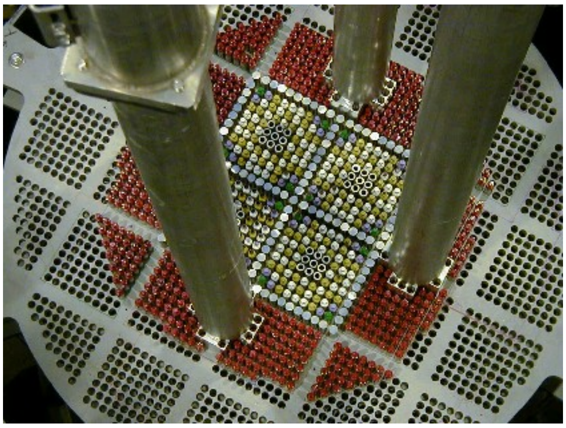


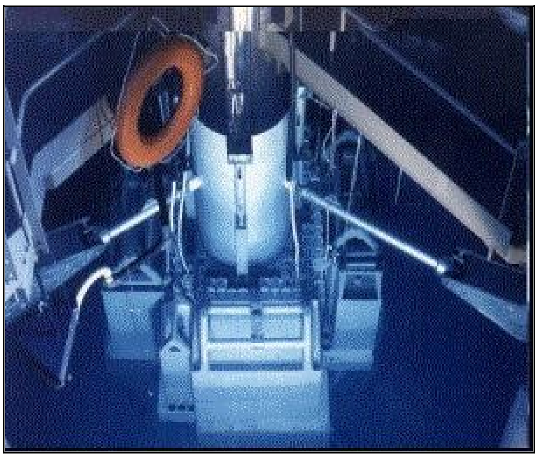
Figure 1: view of EOLE facility (FUBILA Configuration)

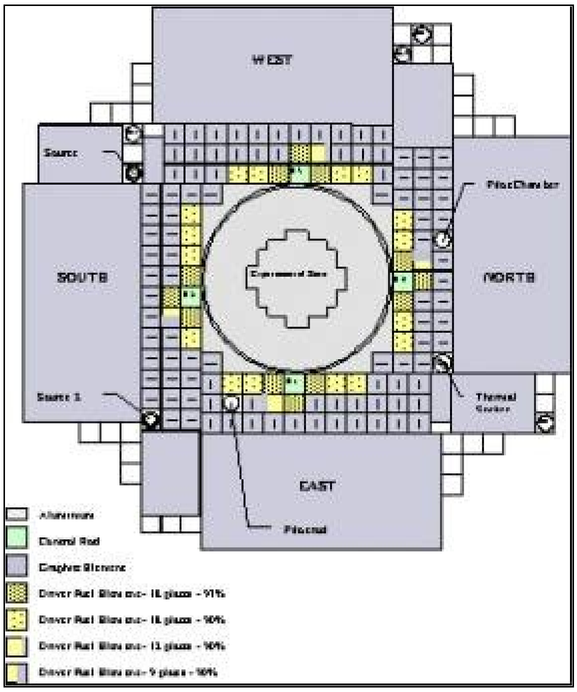
3. THE MINERVE FACILITY

The MINERVE experimental reactor is devoted to neutronic studies of lattices of different reactor types. MINERVE achieved its first criticality in 1959 at the center of Fontenay-aux-Roses (near Paris). The reactor was transferred to CEA-Cadarache in 1977.

MINERVE /**8**/ is a pool type reactor operating at a maximum power of 100 watts. The core is submerged under 3 meters of water and is used as a driver zone for the different experiments located in a central square cavity with a size of about 70 cm by 70 cm. The coupled lattices in this cavity are built such that they can reproduce the neutronic spectra of various reactors

The core is built in a parallelepiped pool of stainless steel containing about 100 m^3 of water. The moderator is distilled water and the cooling is performed by natural convection. The driver zone consists in enriched metallic uranium/aluminium plates cladded with aluminium and gathered in elements of 9, 12, and 18 plates. About 30 elements comprise the driver zone which is surrounded by a graphite reflector. Figures 2 and 3 show the core loading for a standard UO_2 PWR typical configuration (R1-UO_2) of MINERVE facility.





Figures 2 and 3 Core Loading for the R1-UO2 Configuration of MINERVE facility

Several lattices corresponding to different neutron spectra can be built in the central region of the MINERVE Reactor, corresponding to different neutron spectra including an over-moderated UO_2 spectrum (R2-UO2), PWR spectrum (R1-UO2), PWR MOX spectrum (R1-MOX), two epithermal spectra (MORGANE-R, MORGANE-S), moderated fast spectrum (ERMINE), and fast spectrum (ERMINE/COSMO).

The neutron spectra in the experimental zone vary versus the configuration of the lattice. Table 1 includes information on the moderation ratio (*Vm/Vf*) and on the slowing-down density *q* (energy cut-off at 2.77 eV) for the different lattices.

Experimental lattice	*Vm/Vf*	*q*
R1-UO2	1.4	0.60
R2-UO2	1.4	0.80
R1-MOX	1.4	0.55
MORGANE R	0.9	0.35
MORGANE S	0.5	0.32

Table 1 Spectral Information for the experimental lattices in MINERVE

The VALMONT, the HTC, the OCEAN and the OSMOSE programs are being carried out between 2003 and 2011. These programs respectively aim at improving, in different experimental lattices, the accuracy of physics data of high-density UMo/Al fuel, of a majority of the separated heavy nuclides from ^{232}Th to ^{245}Cm appearing during the reactor and the fuel cycle physics. The qualification of capture cross sections of separated isotopes Gd, Hf, Er, Dy, Eu and the reactivity loss per cycle of spent PWR and MOX-PWR fuel samples (→ 70GWd/t) are also performed.

They complete the BURN-UP CREDIT program **/9/** performed between 1994 and 2001 in the same facility, which was dedicated to the validation of the calculation tools accounting for the safety margin issued from the fifteen main non volatile long lived fission products in the criticality studies.

3.1 THE VALMONT PROGRAM

In the framework of the studies dealing with the preliminary safety report for future JHR (Jules Horowitz Reactor) MTR (Material testing Reactor) in CEA Cadarache, a specific experimental program – VALMONT (Validation of Aluminium Molybdenum Uranium fuel for neutronics) **/10/** – was launched in order to study the neutron properties of high-density UMo/Al fuel enriched at 20% in ^{235}U, and thus to qualify neutronics codes.

The VALMONT program **/10/** was performed on the MINERVE facility from November 2003 to March 2004. It was aimed at qualifying the HORUS3D/Neutronics route used for the development of JHR. Thus one of the major tasks was to obtain experimental results with uncertainties as low as possible.

The first part of this programme was devoted to study the reactivity effect of fuel samples - manufactured by CERCA Romans in 2003 - by the oscillation technique. This last one allowed to obtain an accuracy better than 1% on reactivity worth measurements. The goal was also to uncorrelate the effect of the enrichment in ^{235}U, of the density (pressed fuel or sintered fuel), of Aluminium, ^{238}U and Molybdenum, and of the matrix (metallic or oxide).

The second part of VALMONT was dedicated to the characterization of production and absorption effects in UMo/Al fuel enriched at 19.75% in ^{235}U. This was performed loading a specific UMo/Al fuel pin in MINERVE, surrounded by standard UO2 (3% ^{235}U) pins. The results were compared to the ones of measurements in an homogeneous lattice made of UO2 (3% ^{235}U) pins. The following physical parameters were measured : radial power distribution by integral and particular peak gamma-scanning, axial ^{237}Np and ^{235}U fission rates using fission chambers, axial power distribution by integral gamma-scanning, reactivity worth of the UMo/Al fuel pin by reactivity excess measurement, and finally modified conversion factor of ^{238}U by particular peak gamma-spectroscopy.

3.2 THE OSMOSE PROGRAM

The objective of the OSMOSE program**/11/, /12/** is to measure very accurately the integral reaction rates in representative spectra for the actinides dealing with future nuclear system designs and to provide experimental data for improving the basic nuclear data files. These data will support advanced reactors designed for transmutation of waste or Pu burning, sub-critical systems such as found in advanced accelerator applications, and the waste disposal and treatment program in the area of criticality safety. This program is very generic, in the sense that it will measure these reaction rates over a broad range of isotopes and spectra and will be used to provide guidance to all nuclear data programs in the world. These data will provide information valuable to a large number of projects as noted above.

The need for better nuclear data have been stressed by various organizations throughout the world, and results of studies have been published which demonstrate that current data are inadequate for designing the projects under consideration. In particular, a Working Party of the OECD has been concerned with identifying these needs and has produced a detailed High Priority Request List for Nuclear Data. The first step in obtaining better nuclear data consists in measuring accurate integral data and comparing it to integrated energy dependent data: this comparison provides a direct assessment of the effect of deficiencies in the differential data. Several US and international programs have indicated a strong desire to obtain accurate integral reaction rate data for improving the major and minor actinide cross sections. Specifically, these include: ^{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, ^{242}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm, and ^{247}Cm. Data on the major actinides (i.e. ^{235}U, ^{236}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{241}Am) are reasonably well-known and available in the Evaluated Nuclear Data Files - (JEF, JENDL, ENDF-B). However, information on the minor actinides (i.e.^{232}Th, ^{233}U, ^{237}Np, ^{238}Pu, ^{242}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm, and ^{247}Cm) is less well-known and considered to be relatively poor in some cases, having to rely on models and extrapolation of few data points. This is mainly due to the difficulty of obtaining pure samples of sufficient quantity (up to about one gram) to perform reliable reaction rate measurements.

The OSMOSE program aims at providing precise experimental data (integral absorption cross-sections) about heavy nuclides - ^{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. The study of these nuclides is performed on a large range of neutron spectra corresponding to specific experimental lattices (thermal, epithermal, moderated/fast, and fast spectra).

The OSMOSE program will begin in mid-2005 and will continue until 2013

3.3 THE OCEAN and HTC PROGRAM

The MINERVE facility also improves the knowledge of the integral absorption cross sections of new absorbers (OCEAN program) and of the High burn Up area (HTC program) in thermal, epithermal and fast spectra. These studies are used to extend the validity domain of the criticality and radioactivity calculational tools.

The OCEAN and the HTC programs are being carried out between 2003 and 2009. These programs respectively aim at improving, in different experimental lattices, the accuracy of physics data in epithermal, PWR and MOX-PWR spectra of :

- separated isotopes of absorbers:
  ^{155}Gd, ^{157}Gd, Gd nat ^{177}Hf, ^{178}Hf, ^{179}Hf, ^{180}Hf
  ^{166}Er, ^{167}Er, ^{168}Er,^{170}Er ^{160}Dy, ^{161}Dy, ^{162}Dy, ^{163}Dy, ^{164}Dy, ^{151}Eu, ^{153}Eu, Eu nat
- the reactivity loss per cycle of :
  spent UOX-PWR fuel samples (from 50 to 75 GWd/t)
  spent MOX-PWR fuel samples (from 10 to 55 GWd/t).

4. THE MASURCA FACILITY

The MASURCA facility (figure 4) is dedicated to the neutronic studies of fast lattices. It is an airflow cooled fast reactor operating at a maximum power of 5kW and a flux level up to 10^{11} n/cm^2/s. The materials of the core are contained in cylinder rodlets, along with in square platelets. These rodlets or platelets are put into wrapper tubes having a square section (4 x 4 inches) and about 3 meters in height.

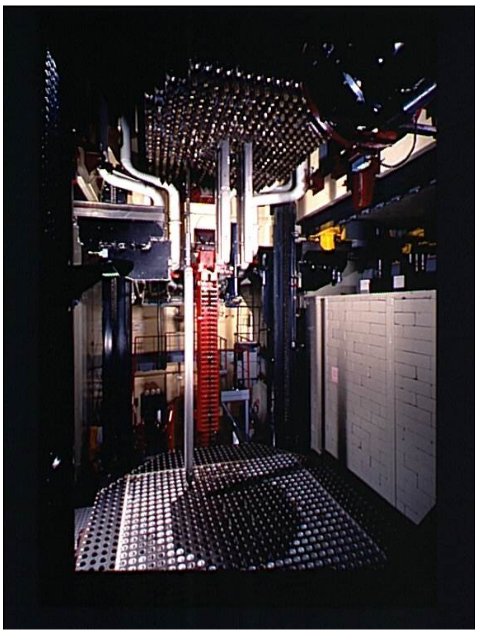


Figure 4 view of MASURCA facility

The reactivity control is fulfilled by absorber rods in varying number depending of core types and sizes. These safety rods (SR) are composed of fuel material in their lower part, so that the homogeneity of the core is kept when the rods are withdrawn. The core is cooled by air. Cores of various dimensions (1000 tubes maximum) can be simulated.

MASURCA had its criticality in 1966. Till the middle of the 90's, programs were devoted to the studies conducted in support of the PHENIX, SUPERPHENIX and EFR (European Fast Reactor) projects. Since, it has been used to test various core concepts that could be used for the transmutation of radioactive wastes.

The COSMO program (1998-1999) aimed to study the Physics of the irradiation of LLFP targets in moderated sub-assembly at the periphery of fast reactors. The effects of various moderators ($^{11}B_4C$, CaH, , ZrH_2), have been investigated. This program helped in particular to the preparation of the ECRIX experiments that have been loaded in the PHENIX reactor in 2004.

Since 2000, the fourth phase of the MUSE program allowed to study configurations with reactivity level representative of those envisaged for future ADS **/13/ /14/ /15/ /16/**.

Mainspring of the MUSE-4 experiments, the GENEPI (Générateur de Neutrons Pulsés Intense) pulsed neutron generator has been built specifically with a view to these experiments. Its main characteristic is to deliver very short pulses (<1µs) with a repetition rate going from some hertz to 4.5kHz. The GENEPI beam guide is horizontally introduced at the core mid-plane and the target is located at the core center (Figure 1, right). Two different neutron sources, produced respectively by D(d,n)^{3}He reactions and D(t,n)^{4}He reactions, have been used. The neutron production rates were respectively found equal to 3.0±0.3 10^4 neutrons per pulse and 3.3±0.3 10^6 neutrons per pulse for a fresh tritium target.

The MUSE-4 series of experiments that ranged from critical to k~0.95 have provided an extremely important set of data to the ADS community. The coupling of neutron sources of different energies (Cf, DD, and DT) allowed us to investigate different regimes of source importance. In spite of measurements very time consuming because of the reactivity levels and time constraints, the MUSE-4 program has produced an impressive amount of data which will help the analysts in their understanding of the behaviour of a sub-critical fast assembly.

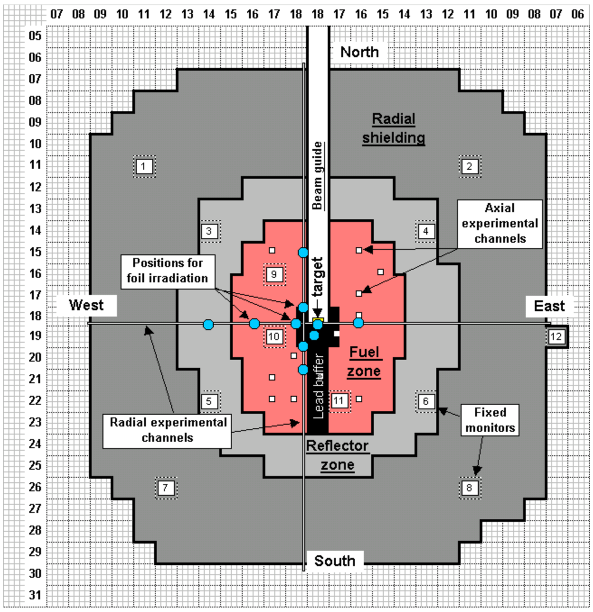


Figure 5 Core Loading for the MUSE4 Configuration

Perhaps more important for the ADS community, the MUSE-4 program has provided a means of testing a variety of reactor physics experimental techniques for subcritical reactor monitoring. Much effort was devoted to dynamic methods. Among the main conclusions, it was found that traditional noise methods (driven by the intrinsic or Cf source) such as the CPSD, Rossi- α and Feynman-α were usable only in the case where the reactivity was close to critical. A Rossi type technique driven by the neutron generator is more promising, but it remains to be seen if these kinds of methods will have a role in the reactivity monitoring of an ADS. At this point, it appears that some form of the PNS method will be the most useful. Studies in MUSE-4 have shown that the so-called "area method" is quite forgiving in relation to other models based on the study of the differential (slope) of the signal (2 or 3 regions). It seems that this method could be used in a cold start-up configuration of an ADS, particularly since small neutron generators are easily obtained. For the day-to-day monitoring however, it appears that some form of a current/power/reactivity relation will be used.

Development of Generation IV nuclear system is one of the major R&D directions in France in the broad area of innovative reactors, fuels, and fuel cycles. The CEA Generation IV reactor physics activities are largely focused on the GFR and notably the development and the validation of calculation tools. In order to support the development of a prototypic GFR, a reactor physics integral experimental program, the ENIGMA program, is being planned in the MASURCA facility. This multi-year, scheduled in 2009, after the refurbishment of the facility will give an important database on generic GFR physics issues by Reference core Characterisation study and central substitutions (C, Si, Zr, degraded Plutonium, absorbers,…), reflectors, streaming studies.

4. CONCLUSION

The critical facilities of the CEA Cadarache are deeply involved in the research program concerning the future of plutonium and wastes management. The experimental programs defined in the EOLE, MINERVE and MASURCA facilities aim at improving the calculation schemes by reducing the uncertainties of the experimental databases for nuclides arising in plutonium and wastes management. They also provide accurate data on innovative systems in terms of new materials (moderating and decoupling materials) and new concepts (ADS, ABWR, GEN IV) involving new fuels and coolant materials.

5. ACKNOWLEGMENTS

The authors of this paper thank all the participants to the experimental program and the industrial partners for their financial support, in particular CNRS, EdF, Framatome, JNES and ANL.

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