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REACTOR PHYSICS PROGRAMME

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Objectives

The Reactor Physics & MYRRHA Research Department gathers expertise in different reactor physics fields namely: neutronics calculations, reactor dosimetry, reactor operation, reactor control, and non-destructive analysis on reactor fuel. This expertise is applied within the Reactor Physics & MYRRHA Research Department own research projects in the VENUS critical facility dealing presently with MOX fuel core physics, in the BR1 reactor dealing with neutron dosimetry calibration, ex-core neutron transport and shielding problems for qualification and validation purposes within an international partnership, or for MYRRHA project aiming at developing a pre-design of an Accelerator Driven System. This expertise is also used in programmes external to the department, such as the pressure vessel steel programme, the BR2 dosimetry, the irradiation experiments preparation and interpretation in the BR2 material testing reactor. *The Reactor Physics programme* aims at developing, improving and maintaining the experimental and theoretical expertise of the department in this domain. In the following sections we are reporting on these activities.

Investigation of Military Plutonium

The dismantling of nuclear warheads in the USA and Russia creates a stockpile of weapon-grade plutonium that should be made inaccessible for future use in nuclear weapons. One of the proposed solutions to diminish these stockpiles is to use the plutonium in commercial power reactors. The envisaged stockpile in the US alone represents an amount about 50 tonnes of weapon-grade plutonium equal to an energy quantity of 10 300 GWd_e , which can be used for peaceful purposes, or the year production of 33 power plants of 1000 MW_e.

Since the neutronic behaviour of weapon-grade plutonium is different from that of civil MOX, it is considered necessary to investigate this neutronic behaviour for the validation of neutron codes.

In the past some experiments have been performed at the VENUS reactor with MOX having a Pu-vector of 96% ^{239}Pu, 4% ^{240}Pu. Based on this experience a UO_2 reference configuration has been loaded in the VENUS in the course of November/December 1999 to investigate the validity of the previous experiment with modern measurement methods and to perform the experiment in a lattice representative for modern PWR's. Neutron-physical parameters will be calculated and compared with the experimental results.

The aim of the programme is to provide data for validating neutron physics codes for weapon-grade plutonium. Basic data for such a validation are in general k_{eff} and local fission rate distribution measurements. Additional data are e.g. spectrum index and delayed neutron fraction beff measurements.

The amount of available rods with 0.7/4.3 weapon-grade plutonium (0.7% ^{235}U, 4.3% weapon-grade plutonium) is very limited (25 rods of 50 cm length). This means that especially a k_{eff} validation is very hard to obtain with these rods, since the difference in k_{eff} is only in the order of several 10's to maximum 100 pcm, while the average neutron code calculates keff with an uncertainty of about 500 pcm.

However, there are also about 400 MOX rods available with 3/1 weapon-grade plutonium. A major drawback of these rods is that they contain only 1% Pu, but due to the large available quantity these rods can provide a valuable benchmark for k_{eff} calculations. They have less value for fission rate distribution measurements.

More space-dependent parameters like the local fission rate distribution can be determined more easily with the 0.7/4.3 rods. Calculations on different configurations [1] have shown that differences in local fission rate distributions between uranium and weapon-grade plutonium are large enough to be useful for code validation. Other calculations have shown that the fission rate distribution measured at midplane of partial* fuel is not significantly different from homogeneously loaded fuel.

Two experimental configurations have been investigated in 1999 and one other will be investigated in 2000, as shown in figures 1a, 1b and 1c. Figure 1a shows the reference UO_2 configuration, figure 1b shows a configuration with a central 24 rod weapon-grade MOX zone, an intermediate zone of 376 3/1 MOX fuel and a UO_2 driver zone, figure 1c shows a partial 52 rod weapon-grade zone and a UO_2 driver zone.

For the UO_2 reference configuration both critical height, fission rate distribution and spectrum indices F5/F9, C8/F9 and F8/F9 have been determined, where F stands for fission, C for capture, 5 for ^{235}U, 8 for ^{238}U and 9 for ^{239}Pu. For the 3/1 configuration depicted in figure 1b only the critical height has been measured.

** partial mens that the central part of the fuel column is filled with 0.7/4.3 MOX fuel pellets (weapon-grade Pu), while the upper and lower partis is filled with 2/3.06MOX fuel pellets (civil Pu).*

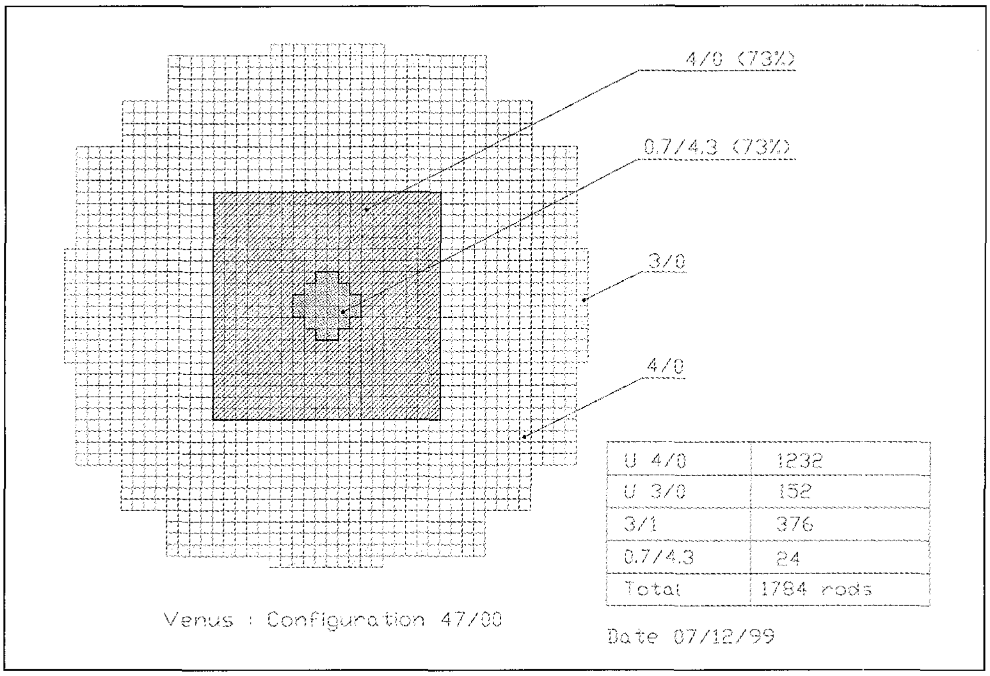

Fig. 1. Typical configuration investigated in the IMP programme

The 52 rod partial weapon-grade MOX configuration is loaded with the purpose to have a benchmark for validating fission rate distribution calculations. This configuration gives in its centre a better approximation of the asymptotic behaviour of the weapon-grade MOX rods compared to the 5x5 configuration. The same is true for spectrum index validation.

The measurements of the first two configurations are in the process of evaluation. A preliminary evaluation of the critical height of the reference UO_2 configuration and the configuration with the central 24 rod weapon-grade MOX zone showed that weapon-grade MOX has a significant influence on the criticality, taken into account that the fissile plutonium content is the same as the ^{235}U content in the reference configuration. The measured reactivity effect was in the order of 800 pcm.

This programme has also been used in the framework of ISTC programme 371.2 to give a practical course on measurement techniques in a critical facility to Russian scientists from IPPE at Obninsk, Russia.

Present criticality safety calculations of irradiated fuel still have to model the fuel as fresh, since no precise experimental confirmation exists of the decrease of reactivity due to accumulated burn-up. The fact that this so-called "burn-up credit" cannot be taken into account has serious economical implications for transport and storage of irradiated fuel.

An international programme for the investigation of the burn-up credit has been initiated (REBUS). The programme aims to establish a neutronic benchmark for reactor physics codes. This benchmark would qualify the codes to perform calculations of the burn-up credit.

The benchmark exercise will investigate the following fuel types with associated burn-up:

- Reference absorber test bundle
- Fresh PWR UO_2 fuel
- Irradiated PWR UO_2 fuel (50 GWd/tM)
- Fresh PWR MOX fuel
- Irradiated PWR MOX fuel (20 GWd/tM)

Reactivity effects will be measured in the critical facility VENUS. The accumulated burn-up of all rods will be measured non-destructively by gamma-spectrometry. Some rods will be analysed destructively with respect to accumulated burn-up, actinides content and TOP-18 fission products (i.e. those fission products that have most implications on the reactivity).

The experimental implementation of the programme will start in 2000.

Neutron and Gamma Calculations Performed for BR2

In order to determine, as in the past, the nuclear characteristics of irradiations projected for, or actually carried out in BR2, neutron and gamma calculations were performed by the Reactor Physics and Myrrha computional team. As most of the irradiation devices are introduced into channels in strongly eccentric positions in the BR2 core (e.g. at the fuel-reflector interface or in the peripheral large H channels), most calculations were performed in two-dimensional geometry with the DORT3.1 multigroup neutron and gamma particle transport (S_N) code. The irradiations in channels inside the BR2 core, i.e. near to the reactor main axis were generally treated with the one-dimensional neutron and gamma particle transport (S_N) code DTF-IV. During the last months of the year, and for the first time, the Monte Carlo code MCNP (version 4B) was applied for the complete detailed modelling of BR2.

The main irradiation devices studied were:

- devices irradiated in the CALLISTO loop for following programmes:
  - BACCHANAL, in which nine water-cooled fuel rods (in a square lattice) are irradiated under PWR conditions. In 1999, no BACCHANAL irradiations were carried out in BR2 due to technological problems, but neutron calculations (with the DORT3.1 code) were made to check some previous work and in preparation of the first cycle 2000. The method used to derive the linear power in each of the nine fuel rods from the experimental thermal balance of the loop, combined with calculated distributions of the fission and gamma heating over the rods, was described and presented at the RRFM'99 meeting;
  - CHIVAS, in which nine steel samples are irradiated. DORT3.1 calculations were carried out in order to determine the fast fluences and the dpa values obtained;
  - THOMOX, in which nine water-cooled fuel rods (in a square lattice) are irradiated under PWR conditions. The rods are of very different types: UO_2, $(U,Pu)O_2$ and $(Th,Pu)O_2$, with various grain sizes and are fabricated according to various methods (sol-gel, MIMAS...). DORT3.1 calculations were started in 1999 in order to determine the irradiation conditions to be expected, as a function of the composition of the fuel rods, of the geometry of some loop components and of the BR2 loading around the irradiation channel;
- loops devised for the irradiation of water-cooled lithium-lead (WCLL) fusion-reactor blanket components in standard channels of BR2: tritium-permeation barrier experiments and double-wall tube experiments. The in-pile irradiation devices and their main performances were presented at the 5th International Symposium on Fusion Nuclear Technology at Rome and indicate promising results (from the neutronic point of view, only one-dimensional survey calculations were performed).
- devices for the irradiation of steel samples in order to obtain the highest possible fast flux and dpa values (MISTRAL programme). The study consisted mainly in investigating the incidence of the type of coolant (NaK or H_2O) on the fast flux and dpa levels. For the rigs studied, these levels decrease by 8 to 10% when water is substituted for NaK.

Another study concerned the refinement of a method to determine the fuel rod fission power based on the derivation by calculation of this fuel rod fission power from the fast and/or thermal flux measured with a dosimeter in a position close to the fuel rod. At the present stage, the calculations were performed in one dimension with DTF-IV, investigating in detail the influence of a whole series of measurement and calculation uncertainties on the results. A presentation of the work carried out was made at the Tenth International Symposium on Reactor Dosimetry, Osaka (Japan), Sept. 12 - 17, 1999.

The effect of neutron and gamma ray spectra on the sensitivity of beta and prompt self-powered neutron detectors (SPNDs) was analysed in the framework of the DOLMEN project of the technology department. Neutron and gamma spectra in typical BR2 channels were considered.

An important irradiation programme started towards the end of 1999, in which novel MTR fuel plates with very high-density meat are being irradiated in BR2. The first irradiation took place in cycle 5/99. The purpose of the irradiations is to investigate the behaviour of these fuel plates under severe reactor operation conditions. The novel fuel plates are inserted in two standard six-tube BR2 fuel elements in the locations normally occupied by the standard outer fuel plates. The irradiation in BR2 was prepared by carrying out detailed neutron Monte Carlo calculations (with the MCNP-4B code) of the whole BR2 core containing the two experimental fuel elements for various positions in the reactor and for various azimuthal orientations of the fuel elements. Comparing the thus determined fission density levels and azimuthal profiles in the new MTR fuel plates irradiated in the various channels allowed the experimenters to choose the most appropriate BR2 channel and the most appropriate fuel element orientation. This was the first time that a Monte Carlo modelling of BR2 was performed. The results of one test case were compared with those of a two-dimensional calculation carried out with the DORT3.1 2-D neutron transport code in (R,Θ) geometry. The results were in good agreement. Various publications will be made in 2000.

Neutron and Gamma Calculations Performed for Other Reactors

A very detailed 3-D MCNP-4B Monte Carlo model of the VENUS reactor was developed in order to replace 2-D deterministic transport calculation methods used up to now for the determination of the design of new experiments and the evaluation of irradiation conditions. The 3-D geometrical model is a nearly exact reproduction of the VENUS facility. In particular, each fuel rod was modelled separately. ENDFB-VI continuous energy cross-section sets were used throughout. Comparisons with reference configurations give a difference of 1% between measured and calculated k-eff values.

Deterministic 1-D transport calculations were performed for the OVERMOX[1] (OVER-moderated MOX) project. This project studies the feasibility of an extension of the VENUS VIP programme to a full MOX core loading. For this purpose, a modification of the VENUS lattice pitch was envisaged and calculations of the k-eff and of the moderator temperature reactivity coefficient as a function of the pitch were performed. This allows the determination of the pitch range of stability of the reactor with respect to the moderator temperature variation.

Updating of Neutron and Gamma Cross-section Libraries

The cross-section processing code system NJOY97 version 95 has been installed and tested on a SUN© Enterprise 2000 multiprocessor machine. In the course of the testing and data processing additional updates, from A. Trkov, IJS, Ljubliana, Slovenia, D.W. Muir, IAEA, Vienna, Austria and H. Wienke, SCK•CEN, were included. Cross-section libraries have been prepared from Jef-2.2 (170 nuclides) and ENDF/B-6.5 data (150 nuclides). The libraries are in Vitamin-B6 multigroup structure (199 neutron and 42 photon groups) as well as in the ACE format, compatible with the N-Particle Monte-Carlo transport code system MCNP-4B. Preliminary results of calculations of the OECD/NEA benchmark for an Accelerator-Driven Minor Actinide Burner[2], performed using MCNP-4B and the JEF-2.2 ACE library showed good agreement with those from other participating groups.

The multigroup data were in the NJOY/GENDF and /MATXS formats. The MATXS data are to be post-processed using the LANL post-processing codes TRANSX-2.15 and BBC[3], which have been installed and tested as well.

A procedure has been established for preparing AMPX master libraries from multigroup data in NJOY/GENDF format[4]. Post-processing of these libraries turned out not to be possible with the Scale-4.4 code system, due to a bug in the latter as came out afterwards, but was possible with Scale-4.3.

Implementation of Reactor Codes and Management of the Unix Workstations

The production of results, from calculations to conclusions, is performed on different dedicated UNIX systems at the Reactor Physics and Myrrha department. A complete system administration is provided to all the computational team members. This service consists in both software and hardware management. Software means all the applications running on the workstations: the operating systems, a wide range of tools and the computer codes. Hardware means the systems used to compute (called servers) and those used to access the servers (called clients). The operating system kernels are configured in order to give high performance computing power.

Before being used, the codes and the nuclear data libraries need to be installed and tested. Most of them need to be recompiled and/or adapted to take as

1. *D. Marloye, "Etude de la Faisabilité d'un Programme Expérimental sur les Cœurs Surmodérés Full-Pu au Réacteur VENUS", Technical Note SCK•CEN SPL-512/95-05, SCK•CEN, 1995.*
2. *Report NEA/NSC/DOC(99)13*
3. *R.E. MacFarlane,, "TRANSX 2: A code for interfacing MATXS cross section libraries to nuclear transport codes", LA-9863-MS*
4. *P. de Leege, "NSLINK-4.2, Code for Preparation of AMPX Master Libraries from NJOY/GENDEF Data", IRI, Delft, Netherlands, unpublished.*

much as possible advantage of the different system architectures and improve the overall performances. Codes or libraries such as BUGLE-96, DOORS 3.2, MCNP-4B (patched), NJOY 97, TRANSX-2, ORIGEN 2.1, SCALE 4.4, SCAMPI, VITAMIN-B6, WIMSD-5B were installed and/or upgraded. After successful optimised (re)compilations, the codes must be tested to verify their numerical stability. Minor modifications in the code sources can be made concerning mainly the input/output (I/O) sequences. Performance and tuning are then the main themes in the daily system administration.

Also, nobody ignores that 1999 was a key year for the computer world. Our systems had to be prepared to be "Y2K compliant". Hardware and software parts were carefully checked and all bugs discovered were eliminated.

Fuel Cycle Studies

In the framework of the study "Supporting Nuclear Data for Advanced MOX Fuels", which constitutes part of the specific programme on Nuclear Fission Safety of the Framework Programme for the European Atomic Energy Community (1994-1998), the neutron calculations, started in 1997 in order to validate the actinide evolution computational methods available, in particular for MOX fuel irradiations in thermal reactors, were rounded off. The main conclusions of the study were already given in the Annual Scientific Report 1998. The final European Atomic Energy Commission report will be published in 2000.

In the framework of the study "Evaluation of Possible Partitioning and Transmutation Strategies and of the Means for Implementing Them", which also constitutes part of the specific programme on Nuclear Fission Safety of the Framework Programme for the European Atomic Energy Community (1994-1998), actinide and fission product evolution calculations of UO_2-PWR and MOX-PWR fuel rods were performed with the codes ORIGEN-2 and SAS2H (which is part of the SCALE4.3 computing system). The results were compared with those from calculations performed by CEA with the CESAR code. For the UO_2 fuel, rather large differences were observed for many nuclides (both between ORIGEN-2 and CESAR and between SAS2H and CESAR); for the MOX fuel, the ORIGEN-2 and the SAS2H results were in general closer to the CESAR results. The final European Atomic Energy Commission report will be published in 2000.

Two papers, related to advanced fuel cycle, were presented at the International Conference on Future Nuclear Systems GLOBAL'99. The first paper discusses scenarios with LWR-UO_2 reactors, LWR-MOX reactors, FRs and ADS systems. The second paper compares FR-MOX and ADS-MOX irradiations as to the consumption of minor actinides (MAs).

A report concerning possible fuel cycles in Belgium was elaborated in the framework of a study in preparation of a national energy debate.

SCK•CEN participated in various CAPRA/CADRA meetings and determined together with CEA the conditions for joining the CAPRA/CADRA project as a full-part member.

Presentations

E. Malambu, Ch. De Raedt, M. Weber, "Assessment of the Linear Power Level in Fuel Rods Irradiated in the CALLISTO Loop in the High Flux Materials Testing Reactor BR2", presented at the 3rd International Topical Meeting "Research Reactor Fuel Management (RRFM)", Bruges, March 28-30, 1999.

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Ch. De Raedt, L. Baetslé, E. Malambu, H. Aït Abderrahim, "Comparative Calculation of FR-MOX and ADS-MOX Irradiations", presented at the International Conference on Future Nuclear Systems, GLOBAL'99, Jackson Hole (Wyo.) August 30-September 2, 1999.

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A. Haghigat, H. Aït Abderrahim, G.E. Sjoden, "VENUS-3 Modeling with PENTRAN," presented at the Tenth International Symposium on Reactor Dosimetry, Osaka (Japan) September 12-17, 1999.

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PH. BENOIT, CH. DE RAEDT, M. DECRETON, "In-Pile Testing of WCLL Double Wall Tubes", presented at the 5th International Symposium on Fusion Nuclear Technology, Rome, September 19-24, 1999.

J. BASSELIER, TH. MALDAGUE, A. RENARD (Belgonucléaire), H. AÏT ABDERRAHIM, S. BODART, PIERRE D'HONDT, K. VAN DER MEER (SCK•CEN), V. RYPAR, C. SVOBODA, I. VASA (Nri), "Critical Experiment with Spent Fuel for Burn-Up Credit Validation (Rebus International Programme," presented at the Sixth International Conference on Nuclear Criticality Safety (ICNC'99), Palais des Congrès, Versailles, France, September 20-24, 1999.

K. VAN DER MEER, B. VERBOOMEN, D. MARLOYE, P. D'HONDT (SCK•CEN), TH. MALDAGUE, J. BASSELIER (Belgonucléaire), "Neutron Code Validation for MOX Fuel with the VENUS Critical Facility", presented at the Sixth International Conference on Nuclear Criticality Safety (ICNC'99), Palais des Congrès, Versailles, France, September 20-24, 1999.

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