STATUS OF THE REBUS-PWR INTERNATIONAL PROGRAMME

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ABSTRACT

The paper presents the REBUS-PWR International Programme, a Burnup Credit research programme based on "integral" reactivity measurements. After a recall of the scope, special attention is devoted to the calculation work as a support of the programme and first comparisons between calculation and experiment are given.

1. INTRODUCTION

Taking into account the loss of reactivity with burnup in criticality studies may result in significative economical benefits as well as in a decrease of radioactivity burden. This concept, known as "burnup credit" (BUC), has motivated nuclear industry to carry out various R&D programmes in that field.

Nuclear actors claiming to implement BUC have indeed to prove to their respective Safety Authorities that :

- they have the ability to determine the depletion and change of composition of the fuel with burnup,
- they can compute the reactivity effects induced by this modification, with reasonable accuracy, based upon a knowledge of the physical data,
- they master an efficient and conservative BUC methodology.

To this end, the international programme REBUS-PWR (Reactivity tests for a direct Evaluation of the Burn-Up credit on Selected irradiated LWR fuel bundles) has been initiated jointly by the Belgian Nuclear Research Center SCK•CEN and BELGONUCLEAIRE. Additionally sponsored by EdF (France), VGB (German nuclear utilities), and NUPEC (Japan), REBUS aims at providing a unique experimental database for validation of reactor physics codes devoted to burnup credit. REBUS addresses two of the here above mentionned points :

- calculation to experiment comparisons for the depletion of the fuel,
- calculation to experiment comparisons for the criticality aspects.

Four major features make REBUS an outstanding programme :

1. The use of a commercial fuel irradiated in a standard way, up to a high burnup. The "integral" approach makes the burnup effect very demonstrative.
2. The irradiated fuel is chemically characterized, in a similar way to the ARIANE [1] programme, but with the difference that the analyses focus on a slightly restricted list of nuclides (actinides, burnup indicators and the most important fission products (FPs) with regard to neutron absorption) to be performed by the SCK•CEN.
3. The antireactivity induced by the burnup, assessed through critical configurations, is foreseen to be large enough to reduce the relative importance of all uncertainty sources.
4. The critical configurations are made as simple as possible so that they provide a valuable database for code benchmarking.

The present paper is organized with four main parts. The first paragraph recalls the scope of REBUS-PWR, or what has to be done. The second one deals with the neutronic calculations as a support of the programme. The third paragraph presents the status of REBUS-PWR, or what is effectively done at the time of writing, including some comparisons between experiment and calculation.

2. SCOPE OF REBUS-PWR

The scope of REBUS-PWR has been recently described into details [2], so that only the main points are recalled here. The scope includes the study of five critical configurations as well as non-destructive and destructive examinations. The critical configurations are to be loaded in the VENUS facility [2, 3]. They consist of a driver zone surrounding a central test bundle, which is successively composed of : 1. The same UO_2 fuel rods as in the driver zone (reference case), 2. Fresh commercial UO_2 fuel rods, 3. Irradiated commercial UO_2 fuel rods (54 GWd/t_M), 4. Fresh MOX fuel rods and 5. Irradiated MOX fuel rods (20 GWd/t_M).

2.1 HOT CELL WORK

A refabrication of the commercial spent fuel rods from 4 meter rods into 1 meter rodlets (2 test rodlets per commercial rod) is to be executed in the SCK•CEN hot cell laboratory. Afterwards the rodlets are cleaned thoroughly, since the contamination level of the VENUS reactor has to remain below very low specified limits. Also the BR3 spent fuel rods have to be cleaned, but in this case no refabrication is needed as the original length is already 1 meter. After cleaning the rodlets will be assembled in the experimental 7×7 bundle (paragraph 2.3).

2.2 FUEL CHARACTERIZATION

As already said, one of the features of REBUS is to provide both reactivity and the related isotopic inventory information. The characterization of the spent fuel is performed in two steps. The fresh fuel content is well-documented during fabrication.

The first step is non-destructive and is performed before the reactivity measurements. All spent fuel rods are measured by gross gamma scanning in order to get the axial burnup profile. One specific rod is investigated by gamma spectrometry, together with a well-qualified calibration source, to determine the ^{137}Cs content and in this way the burnup of the rod. The combination of the gamma spectrometry and the gross gamma scans gives a good picture of the burnup of all rods. This first step is necessary to verify that the selected rods have a similar burnup and to align the active fuel columns.

The second step is destructive and is carried out on one selected rod per bundle. It aims at determining both the actinides content (U, Np, Pu, Am, Cm), some burnup indicators (^{137}Cs, ^{144}Ce, Nd) and the 19 most important (TOP-19) fission products (Sm, Eu, Tc, Mo, Ru, Rh, Ag, Gd, Pd, Nd, Cs) with respect to neutron absorption, representing more than 80 % of the neutron absorption in the spent fuel. The sample for this destructive radiochemical assay is taken from the same rod, on which gamma spectrometry has been performed.

2.3 CRITICAL EXPERIMENTS PERFORMED AT VENUS

Five fuel bundles (Figure 1) will be investigated in the framework of the REBUS-PWR programme :

- Bundle 1 : reference 3.3 w/o enriched UO_2 fuel
- Bundle 2 : fresh commercial PWR UO_2 fuel, manufactured by Framatome ANP (formerly SIEMENS)
- Bundle 3 : irradiated commercial PWR UO_2 fuel (54 GWd/tM + 5 years cooling), originating from Neckarwestheim NPP and belonging to GKN (Germany)
- Bundle 4 : fresh PWR MOX fuel, originating from the BR3, an experimental Belgian PWR
- Bundle 5 : irradiated PWR MOX fuel (20 GWd/tM + 15 years cooling), also from the BR3 reactor.

All test bundles will be loaded as a 7×7 fuel assembly in the center of a UO_2 fuel driver zone. The 7×7 assembly is chosen because calculations show that such an assembly will result in a reactivity effect that is large enough for benchmark purposes (~ 1500-2000 pcm).

The driver zone is mainly made of 3.3 w/o enriched UO_2 fuel rods (external dimensions 23×23) and two additional rows of 4.0 w/o enriched UO_2 fuel rods, leading to external dimensions of 27×27 rods. The same driver zone is to used for the five critical configurations of the REBUS-PWR programme.

The reference 3.3 w/o enriched UO_2 fuel bundle consists of the same 3.3 w/o enriched UO_2 fuel as there is in the driver zone. Its purpose is to calibrate the k_{eff} calculation.

The fresh commercial fuel bundle consists of a 5×5 fuel assembly. The fuel is 3.8 w/o enriched UO_2 fuel, fabricated at Framatome ANP (formerly SIEMENS), Germany. It is the original composition of the irradiated commercial fuel.
The irradiated commercial fuel bundle is the same as the fresh bundle, as it is obvious for experimental reasons (clean comparison of the fresh and irradiated bundles).

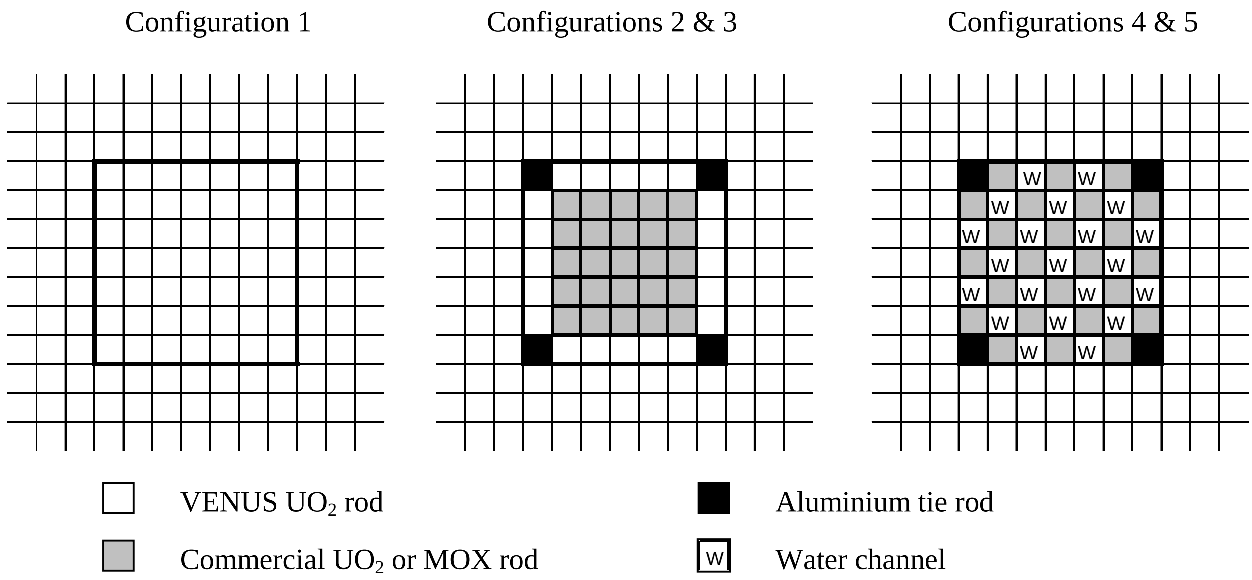

Figure 1. Centres of the critical configurations to be loaded in VENUS in the frame of REBUS-PWR

The fresh BR3 MOX fuel bundle consists of 24 fuel rods set in an overmoderated way (one cell over two). This overmoderation aims at maximizing the antireactivity effect induced by the burnup. The fuel is 6.9 w/o enriched fissile MOX fuel, fabricated at BELGONUCLEAIRE, Belgium. It is the original composition of the irradiated BR3 fuel. The irradiated PWR MOX fuel originates from the BR3 reactor, Belgium and is provided by SCK•CEN.

The commercial UO_2 fuel rods were irradiated inside one 18×18 PWR assembly (see also Figure 3) including burnable poisons UO_2-Gd_2O_3. The 16 central rods are chosen to build the REBUS rodlets, in order to get a burnup distribution as homogeneous as possible inside the bundle. The REBUS MOX rods were irradiated inside various BR3 assemblies containing each 28 rods and arranged within a square pitch. The assemblies are although arranged within a hexagonal pitch inside the BR3 core vessel.

The reactivity effect is measured by loading the different bundles in the centre of the driver zone and measuring each time the critical water level and the reactivity effect of a change of the water level. From these measurements the burnup induced reactivity effect can be estimated (paragraph 3.6).

In addition of the reactivity measurements, fission rate and flux distributions at the main axes will be measured. Due to the impossibility of measuring this parameter in the spent fuel assembly, wire activation measurements will be performed in this assembly to get the neutron fluxes. Table I gives a practical summary of the measurements to be performed at VENUS.

Table I. Overview of the measurements in the different configurations.

Configuration	Critical level	Reactivity effect	Fission rate distribution	Flux distribution
Reference UO_2	X	X		
Fresh UO_2	X	X	X	X
Irradiated UO_2	X	X	in driver zone	X
Fresh MOX	X	X	X	X
Irradiated MOX	X	X	in driver zone	X

3. NEUTRONIC CALCULATIONS IN SUPPORT OF REBUS-PWR

Several kinds of calculations must be performed in support of the experimental programme. The calculations are generally carried out following two steps :

- orientation calculations, aiming at determining the layout of the REBUS-PWR configurations,
- predetermination calculations aiming at foreseeing the critical water level and giving a set of physical parameters needed before and after the experiments.

The methodologies used in the two steps are about the same and one presents hereafter a complete set of calculation work for a couple (fresh/burnt) of critical configurations.

3.1 THE CODES

WIMS8a [4] modular codes are the main neutronic calculation tools used in this context. The WIMS codes can be used « stand alone » or coupled to other codes, such as DANTSYS [5] and KENO-Va [6], with the help of a home-made interface software.

The WIMS8a codes are fed with their own WIMS'97 library based on JEF2.2. This cross sections library is composed of 172 energy groups and contains approximately 80 fission products, representing more than 99 % of the neutronic absorption.

Table II gives a summary of the computation schemes and the corresponding physical parameters they address, as is discussed here under.

3.2 DEPLETION CALCULATIONS

Depletion calculations are performed in order to get the detailed isotopic inventory of the spent fuel to be loaded in VENUS (Configurations #3 and #5) and consequently the related reactivity effect compared with the fresh fuel bundle.

The data coming from the irradiation follow-up of the selected rods are firstly handled so that we determine the average burnup for one rod, representative of the 24 (MOX case) or 25 (UO_2 case) rods that constitute the test bundle.

Table II. Summary of the codes and methods used in the calulations for REBUS-PWR

Parameter	Code	Method
Spent fuel isotopic inventory	WIMS8a (WIMS'97 fed, 172 gr.)	Transport + burnup equations
Critical axial buckling B_Z^2 Delayed neutron fraction β and importance factor I Relative fraction of delayed neutrons a_j and decay constants λ_j Prompt neutron lifetime ℓ	WIMS8a (WIMS'97 fed, 172 gr.) (LWRWIMS for checking)	Transport characteristics Fission rate contributions from the actinides and delayed neutron data
Cross sections preparation	WIMS8a	8 energy groups, P_0 Legendre expansion
Critical water level H_c Reactivity of the water level $\partial\rho/\partial H$	KENO-Va THREEDANT	Monte Carlo 2000 $\times$ 2000 Transport S_4
Total extrapolated length λ	/	Buckling formula

Figure 2 shows the axial burnup profile of these defined rods. The axial burnup profile of the UO_2 rodlets refabricated in the hot cell laboratory is foreseen to be quite flat as the median 2 meters section is chosen (2 test rodlets per 1 commercial rod). The representative UO_2 spent fuel rod is considered to be burnt up to 54.0 GWd/t_M, which is the burnup average over the axial positions of the 24 selected rods. This average burnup is affected by a burnup dispersion of maximum 1.5 %.

The axial burnup profile of the MOX rods are characterized by a strong cosine burnup profile as the orginal BR3 reactor contained 1 meter active fuel rods. The representative MOX spent fuel rod is considered to be burnt up to 20.3 GWd/t_M (axial average). The dispersion of the burnups reaches up to 14 % (less homogeneous test bundle) due to the various assemblies from which the rods are extracted.

Figure 3 shows the 2D geometrical models used to simulate the irradiation of the fuels : a special macrocell representing a BR3 assembly comprised between two moderator tubes for the MOX rods and a single 18×18 PWR assembly for the UO_2 rods. The representative REBUS rods, indicated by the shaded cells, are supposed to be irradiated up to the mentionned burnup. A typical calculation sequence is composed of the following WIMS steps :

- a multicell collision probability calculation,
- a flux calculation in a detailed energy groups structure,
- an energy condensation from 172 to a chosen number of groups (usually 6 for PWR irradiations),
- the main transport « characteristics » calculation (CACTUS module),
- the integration of the burnup equations.

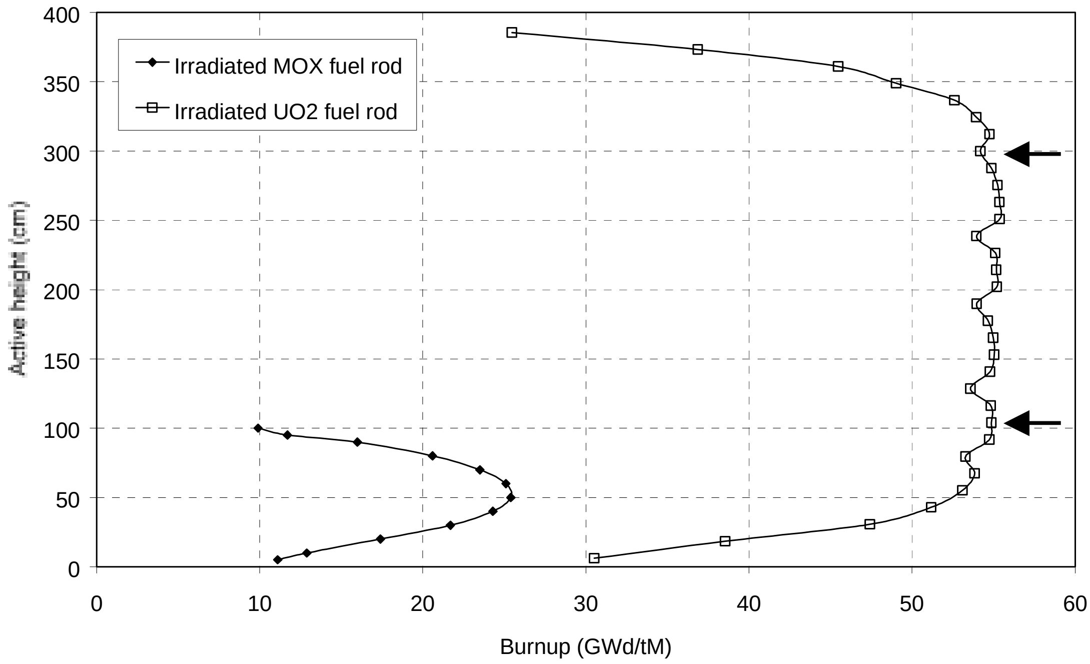

Figure 2. Axial burnup profile of one BR3 MOX and one GKN UO_2 fuel rods. Notice that only the central 2 meters section (see arrows) of UO_2 rods is to be used.

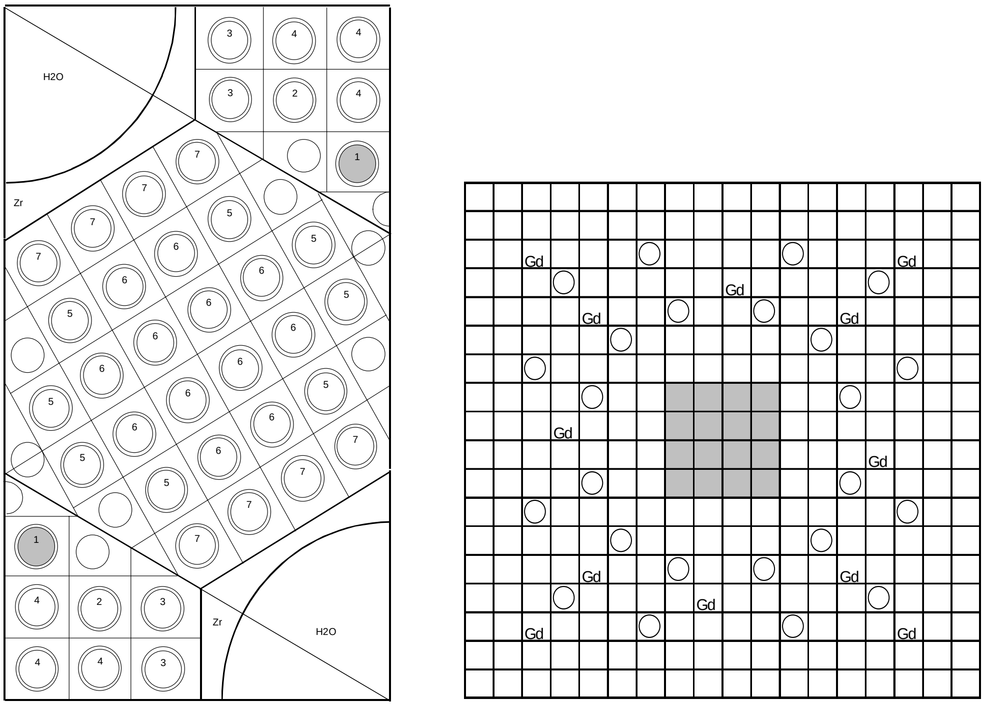

Figure 3. Geometrical model for the depletion calculation. On the left, a macro-assembly model for the representative MOX rod (indicated by the shaded cell) to be depleted up to 20.3 GWd/t_M. On the right, a 18×18 assembly whose central 16 rods indicate the representative rod to be depleted up to 54.0 GWd/t_M.

3.3 VENUS 2D CALCULATIONS

VENUS 2D calculations (WIMS8a) are performed in order to determine a wide set of physical parameters :

- the critical axial buckling B_z^2 (cm^{-2}),
- the fission rate distribution between the different actinides for typical fuel rods,
- the radial pin power distribution,
- the effective delayed neutron fraction β_{eff} ,
- the relative fraction of delayed neutrons and decay constants, within the usual 6-groups structure,
- the prompt neutron lifetime ℓ .

The calculation scheme is nearly the same as for depletion calculations, but the geometrical model represents the flooded fuel region of the REBUS-PWR configuration. In the computation, WIMS8a takes advantage of the 1/8th reflective symmetry.

The critical axial buckling B_z^2 (cm^{-2}) is obtained so that extrapolated length can be further determined (see later).

Fission rate distribution between the different actinides is needed to interpret the fuel rod activity and to provide an experimental pin power distribution. Indeed, one has to take into account the specific yield of the actinides, for the different rod types loaded in VENUS.

The total delayed neutron fraction is determined from the repartition of fissions per isotope *i* and the total neutron source as :

$$\beta = \sum_i \beta_i \tag{1}$$

with
$$\beta_i = \frac{\int \nu_i^\beta \, \Sigma_f^i \Phi \, dV}{\int \nu \, \Sigma_f \Phi \, dV} = \nu_i^\beta \frac{\int \Sigma_f^i \Phi \, dV}{\int \nu \, \Sigma_f \Phi \, dV} , \tag{2}$$

where β_i is the fraction of delayed neutrons due to fissions of isotope *i*, $\int \Sigma_f^i \Phi \, dV$ is the fission rate of isotope *i* (WIMS output) in the configuration and $\int \nu \, \Sigma_f \Phi \, dV \beta_i$ is the total neutron source (WIMS output). The number of delayed neutrons produced after one fission ν^β_i for isotope *i* is taken from [7, 8]. In order to get the effective delayed neutron fraction β_{eff}, one has still to correct the effectiveness of the delayed neutrons (lower energy spectrum than that of prompt neutrons) in the neutron multiplication process, by a so-called "importance factor" :

$$\beta_{eff} = \beta \times I , \tag{3}$$

with
$$I = k_{eff}^{delayed} \big/ k_{eff}^{prompt} . \tag{4}$$

This is achieved by using either LWRWIMS (fed by an older library with 69 groups) in which the neutron emission spectrum is manually modified or a development version of WIMS8a, that allows calculations directly fed with a delayed neutron spectrum (172 groups). A more accurate method to determine β_{eff}, recommended in [9, 10], is thought to be tested in the future.

Once the parameters β_i obtained, the distribution (fractions a_j) of delayed neutrons among the usual 6-groups and the related decay constants λ_j are calculated throught the relations [11] :

$$a_j = \frac{\sum_i a_{ij} \ \beta_i}{\beta} \text{ and } \lambda_j = \frac{\beta}{\sum_i \frac{\beta_i}{\lambda_{ij}}} \ , \tag{5}$$

where the letter *j* is the index of the group of delayed neutrons and *i* refers to the isotope *i*. Finally the prompt neutron lifetime ℓ is deduced through the simulation of a weak and uniform poisoning of the core, by a material whose capture cross section is inversely proportional to the neutron speed. The two-group perturbation theory allows one to determine ℓ by the ratio between the absorption rate of the poison material and the total absorption rate of the core.

The set of kinetic parameters is then used along with the measured period *T* of the core to determine the reactivity of the core through the Nordheim equation :

$$\rho = \frac{\ell}{\ell + T} + \frac{T\beta_{eff}}{\ell + T} \times \sum_{j=1}^{6} \frac{a_j}{1 + \lambda_j T} \ . \tag{6}$$

3.4 CROSS SECTIONS PREPARATION

After the main transport 2D calculation, the cross sections of the fuel, cladding and moderator are smeared together to get equivalent cell cross sections. This permits to build the 3D geometrical model by means of elementary cells characterized by their respective set of cross sections.

Other 2D WIMS8a calculations are carried out to prepare the cross sections of the various materials filling the 3D model, such as structural materials, blankets, plexiglas plugs, etc.

In order to keep the 3D calculation time within reasonnable bounds, the cross sections are so far condensed into 8 energy groups and approximated by P_0 Legendre expansion. The 16 groups P_1 condensation, usually adopted for criticality studies in BELGONUCLEAIRE, will be used the final stage of the work.

3.5 VENUS 3D CALCULATIONS

Geometrical model

Radially, the geometrical model is the same than the one adopted for the 2D WIMS8a calculations.

Axially (see Figure 4), the REBUS configuration is approximated by means of a "6 layers" model. From bottom to top, one sees :
- the far water reflector,
- the near water reflector (mixed with structural material),
- the bottom grid,

- the bottom plexiglas rod reflectors + end plugs,
- the active fuel region (flooded and non-flooded fuel rods, separated by the critical water level H_c),
- the top plexiglas rod reflectors.

Only 1/4th of the core is radially represented as one takes advantage of reflective symmetries. Top and bottom boundary conditions are taken as void.

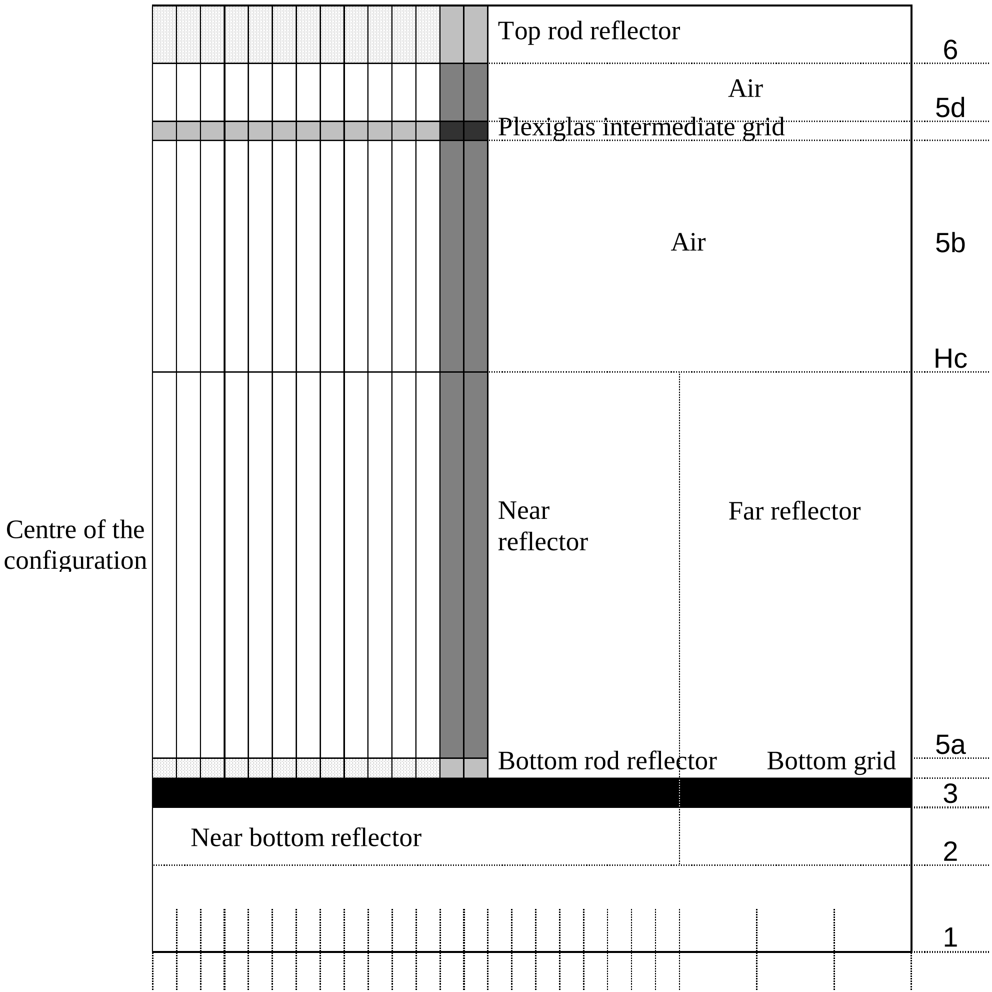

Figure 4. XZ view of the 3D model of the REBUS configuration #1 (reference case). The number of the axial layer is explained in the text. Layer 5 is subdivided into four regions in account of the interface flooded/non-flooded fuels and of the intermediate plexiglas grid.

Calculated parameters

Both the deterministic THREEDANT and statistical KENO-Va codes are used for 3D calculations. THREEDANT is the 3D module of the DANTSYS package. It is a Sn transport calculation and S_4 approximation is chosen here. KENO-Va is a Monte Carlo code and the calculations are here performed with a statistics of 2000 neutrons per generation $\times$ 2000 generations. The geometrical model is identical from one code to the other.

The critical height H_c of the flooded fuel region is obtained through few calculations, from which we also deduce the reactivity of the water level $\partial\rho/\partial H$.

Once the critical height obtained, the extrapolated length λ is deduced by inverting the relation:

$$B_z^2 = \frac{\pi^2}{(H_c+\lambda)^2}, \tag{7}$$

where B_z^2 has been computed by the 2D WIMS8a sequence.

3.6 REACTIVITY EFFECT CALCULATION

In the orientation calculations step, reactivity effects are simply obtained throught the substitution of the fresh bundle by the burnt bundle. 2D WIMS transport calculations, representing the flooded region of the fuel, give both the k-effective of the fresh and the burnt configurations, from which the Δk_{eff} is obtained. In that way we determine approximate reactivity effects induced by burnup of ~ 1600 and ~ 2200 pcm, respectively for the couples of critical configurations incorporating the BR3 MOX fuel and the GKN UO_2 fuel. Such high reactivity effects lead to critical water level differences up to ~ 25 cm.

In the next step, reactivity effects (measured and calculated) between a burnt and a fresh configurations can be evaluated by the simple relation :

$$\Delta\rho = \overline{\left(\frac{\partial\rho}{\partial H}\right)} \times \Delta H_c , \tag{8}$$

taking $\overline{\left(\frac{\partial\rho}{\partial H}\right)}$ as the average water level reactivity between the two configurations. However, due to the large difference in critical height between the couple of configurations, the use of an integration, between the two critical heights H_{c1} and H_{c2}, should be recommended :

$$\Delta\rho = \int_{Hc1}^{Hc2} \frac{\partial\rho}{\partial H}\, dH , \tag{9}$$

using the $\frac{\partial\rho}{\partial H} = \alpha \times H^{-3}$ behaviour of the water level reactivity, where α may vary slightly between the fresh and the burnt configurations. Provisional calculations performed on the commercial UO_2 fuel, show that the two approaches (Eqs. 8-9) lead to reactivity effects that can be different up to 17 %, which is really significative. The calculations also show that the α parameter should not differ more than 5 % in the worst case, thanks to the constant driver zone.

4. PRESENT RESULTS OF REBUS-PWR

4.1 CHARACTERIZATION OF FUEL RODS

The first experimental results of the REBUS-PWR programme deal mainly with the characterization of the fuel. The fresh fuel rods originating from VENUS and BR3 (driver zone, reference bundle and MOX fuel) have already been used in previous benchmark programmes and their characteristics are well-documented. Additionally, the VENUS rods have been elongated from 50 cm to 1 m and during this elongation we took the opportunity to perform some extra measurements with respect to the cladding inner and outer diameter and the position of the lower end plug.

The irradiated MOX fuel from the BR3 reactor has been examined in the hot cell laboratory for gross-gamma scanning (see Figure 5), burnup by gamma-spectrometry, profilometry and fuel column length [2].

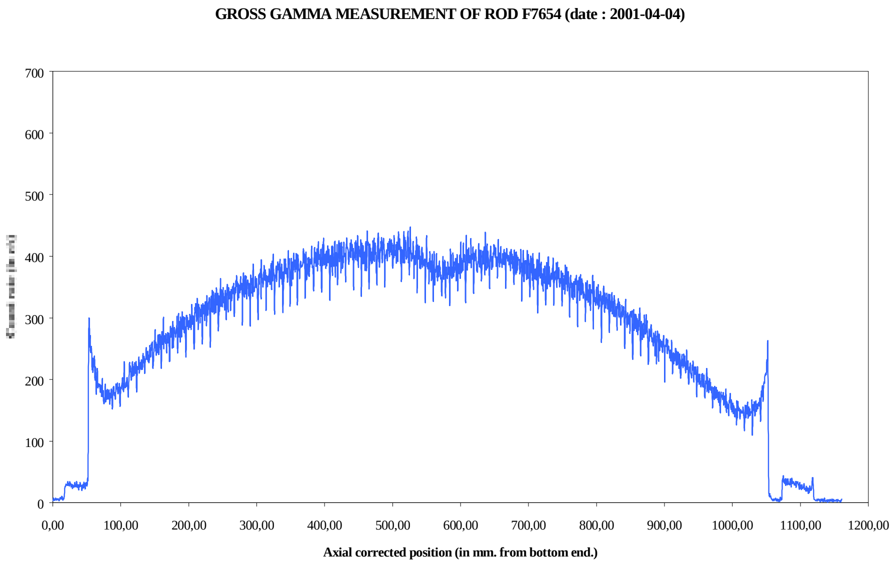

Figure 5. Example of a gross-gamma scan on an irradiated BR3 MOX fuel rod.

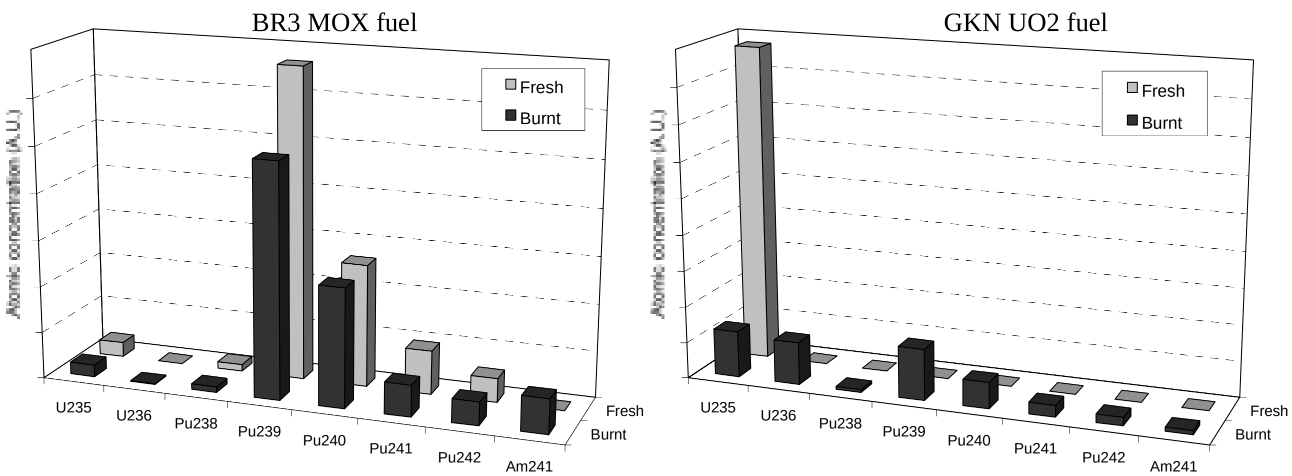

Figure 6. Comparison of the actinides vector of the couples fresh / irradiated fuels.

The gross-gamma scan results show for certain MOX rods a rather high degree of Cs migration, indicating a high linear power during irradiation. Since this Cs migration behaviour is not similar for all rods, we have to make some adjustments with respect to the gamma self-absorption for the 660 keV gamma peak of ^{137}Cs in order to compensate the differences.

Comparison of the axially averaged burnups obtained by gross-gamma scans and by core calculation indicates that the latter slightly overestimates the rod burnup, with an average trend of 3 %.

Depletion calculations performed by the WIMS8a codes (point 3.2) provide the isotopic spent fuel inventory of the representative irradiated REBUS rod (see Figure 6). This composition will be used in future calculations of the VENUS configurations.

4.2 EXPERIMENTS IN THE CRITICAL FACILITY VENUS

At the time of writing, experiments in the critical facility VENUS have been performed on the configurations #1 (reference case) and #4 (fresh BR3 MOX case), but fission rate and flux distributions have still to be analysed. Predetermination calculations, as described in paragraphs 3.3 to 3.5, have thus preceded the experimental work, providing a set of calculated physical parameters (reactivity, kinetic and flux). However those parameters may not be reported here and only comparisons between calculated and measured values are discussed here under.

Table III gives some comparisons for what concerns the critical water level and the reactivity of water level.

Table III. (C-E)/E between calculated and measured reactivity parameters

Configuration	(C-E)/E k_{eff} (pcm)	(C-E)/E $\partial\rho/\partial H$ (%)
Reference (#1)	- 164	-10.3
Fresh BR3 MOX (#4)	+ 263	+ 9.0

Results for the critical water level H_c, obtained with KENO-Va and THREEDANT, are in complete agreement between each other, as the 3D model is exactly the same. However the reactivity effect of the water level is better calculated by the deterministic THREEDANT code than by the statistic code KENO, due to the statistical uncertainty.
We have tested the 16 groups P_1 condensation, for which we observe a decrease of reactivity that can reach ~ 200 pcm of reactivity. This is not so surprising since the P_1 approximation increases the neutron leakage which is not negligible for a such a small core.

4.3 FUTURE STEPS OF THE REBUS-PWR PROGRAMME

Assessments against experimental fission rate and flux distributions (radial and axial directions) for this fresh BR3 MOX configuration, not yet available, could be presented during the conference.

Transport of the fresh GKN UO_2 fuel is foreseen for june 2002, so that configuration #2 could be measured in september 2002 and first comparisons of the critical water level and reactivity of water level could also be presented at the conference.

The critical configurations #3 and #5, incorporating the irradiated bundle, will be measured afterwards. For the irradiated BR3 MOX bundle, the burnup distribution of the test rods required that a special calculation study be performed on the arrangement of these rods inside the bundle, in order to (i) maximize the reactivity effect induced by the burnup and (ii) the optimize the 8^{th} fold reflective symmetry of the bundle. Calculations, simulating either the loading of the highest burnup rods at the centre of the bundle or at its periphery, show that the reactivity difference does not exceed 15 pcm, in favor of the highest burnup rods at the center. The loading of the irradiated BR3 MOX bundle will then be determined on the basis of the gross-gamma scans that reflect the real burnup of the rods.

5. CONCLUSIONS

The REBUS programme will provide an experimental benchmark for burnup credit, taking into account both fissile isotopes depletion and the production of neutron absorbing fission products.

The paper presents the features of the REBUS-PWR programme :

1. an "integral" approach that makes the burnup effect very demonstrative,
2. radiochemical analyses of the actinides content and of the TOP-19 FPs content,
3. large antireactivity induced by the burnup, assessed through critical configurations,
4. simple critical configurations, valuable database for reactor physics code benchmarking.

Although very provisional, comparisons between calculation and measurement show that the remaining discrepancies stay sufficiently low, so that REBUS should indeed provide a clear evidence of the burnup induced reactivity effect.

Possible extensions of the REBUS-PWR programme considering spent fuel, are in preparation, such as REBUS-BWR as well as experiments with high burnup UO_2 and MOX fuels.

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