Nuclear Energy Agency Committee on Reactor Physics

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SUMMARY OF CURRENT NEACRP VIEWS ON FAST REACTOR BREEDING ASSESSMENT

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I - PRESENT BREEDING GAINS FOR OXIDE FUELLED LMFBR

There are different definitions of breeding gain in use. In the simplest way, the global breeding gain (GBG) can be defined as the net equivalent 239 plutonium balance in atoms in the reactor per fission in the reactor. According to the reactor regions where the Pu balance is made, the GBG can be divided in internal breeding gain (IBG) for the core zones or in external breeding gain (EBG) for the blankets, with the obvious relation - GBG = IBG + EBG.

Dealing only in the first part of this summary with mixed oxide-fuelled LFMBR in classical two zones realistic concepts, it is generally agreed that GBG can vary between 0.10 up to 0.35 for power levels varying from 250 to 2000 MWe, depending on the near-term or more advanced oxide fuels considered . These figures must be considered cautiously. They represent only orders of magnitude than can vary versus many parameters : fuel diameter and volumic percentage, steel content, Pu isotopic composition, burn-up, ... The influence of design parameters on GBG values will not be analysed here.

Relative typical contributions of core and blankets to GBG are indicated in the following table for there power levels and realistic plants :

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P_{MWe}	250	500	1200
IBG	- 0.44	- 0.32	- 0.17
EBG	+ 0.56	+ 0.48	+ 0.41
GBG	+ 0.12	+ 0.16	+ 0.24

The loss on EBG when the power level increases is overcompensated by the improvement on IBG. However some studies have shown, in specific cases, larger values of GBG with smaller cores.

The global breeding gain plays a major role in the fuel cycle optimisation and the reactor operation : it defines the net Pu balance per year in the reactor and the corresponding Pu benefit. Doubling time is perhaps of greater significance as a figure of merit to the growth of the LMFBR industry. Since doubling time involves factors with sizable uncertainties, such as number of fuel batches, frequency of refueling, reprocessing and fabrication times and fuel cycle materials not immediately recycles, it cannot be precisely predicted at present. The doubling time, currently used to select optimal concepts, is inversely proportional to GBG. One of the other most useful parameters in assessing the breeding properties of a particular design of fast reactor is an estimate of the total uranium ore requirement per GWe ; that parameter is a function not only of the breeding gain, but also for example of the plutonium inventory and of the out-of pile characteristics. It is worth mentionning that for realistic breeder designs, the yearly natural consumption is about 1,4 tonne per GWe although for standard LWRs this consumption is about 150 t per GWe. Furthermore the IBG is directly connected to the reactivity loss per cycle.

As a whole, the absolute requested uncertainties (2 s.d) generally accepted on breeding gains are :

$\pm$ 0.03 for GBG

$\pm$ 0.02 for IBG

that means $\simeq \pm$ 0.02 for EBG.

These values have to be considered for a reactor for which the enrichments are known : calculations show that increasing average enrichment by 2 % and making no compensating change to maintain criticality results in a reduction of 0,03 in GBG.

<u>NOTA</u> : The breeding ratio, still used sometimes instead of breeding gain, corresponds to the ratio of the fissile atoms built in the reactor to the fissile atoms burnt in the reactor. This parameter first has the main disadvantage to give the same weight to each isotope, second is not really useful to calculate the ^{239}Pu equivalent production by year.

II - MAIN SOURCE OF UNCERTAINTIES

The expression for GBG is the following one :

$$GBG = \frac{\sum_i \left[C_{i-1} - (C_i + F_i)\right] w_i}{\sum_i F_i} \quad ; \quad [a]$$

Where : - C_i, F_i are respectively the capture rate and the fission rate of the isotope integrated over the whole reactor.

- w_i is the weight of the isotope i relative to ^{239}Pu :

$$w_i = \frac{\overline{\sigma}_i^+ - \overline{\sigma}_8^+}{\overline{\sigma}_9^+ - \overline{\sigma}_8^+} \qquad \overline{\sigma}_i^+ = \overline{\nu\sigma_{f\,i}} - \overline{\sigma_{a\,i}}$$

$$w\,(^{239}Pu) = 1 \qquad w\,(^{238}U) = 0$$

- the summation applies to all heavy isotopes.

A detailed analysis of the relative importance of the various sources of uncertainties can be performed through the well-known generalized perturbation theory suggested by USACHEV in 0 or 1D. However, taking into account that for standard concepts, ^{239}Pu fissions represent about 85 - 90 % of the total fission rate, it appears clearly from [a] that the major contributions to GBG come firstly from both ratios C_{238U}/F_{239Pu} and $\alpha_9 = (C_{239Pu} / F_{239Pu})$, secondly from the ratio C_{240Pu} / F_{239Pu} : The orders of magnitude of the relative sensitivities of the GBG to these parameters are given in the following table for a standard 1200 MWe using gaz graphite plutonium and arbitrary uncertainties, without any adjustement of enrichment to maintain criticality :

Parameters	Uncertainty	GBG
C8/F9	± 5 %	≃ ± 0.06
α 9	± 10 %	≃ ± 0.02
Co/F9	≃ 30 %	≃ ± 0.015

Taking only into account the uncertainties on evaluated nuclear data coming from nuclear physics, the corresponding uncertainties on GBG can be as large as ± 0.1 (2 s.d), greatly outside the limits of the design requests. But in that latter case, where only differential nuclear data are considered, without the constraint of a critical system, then nu and alpha for 239 Pu are the most important parameters in assessing breeding.

III - REACTOR PHYSICS STUDIES APPLIED TO GBG

To reduce uncertainties on GBG, reactor studies are performed in three directions depending on the various countries :

. mock-up approach

. parametric analysis of the neutron balance and data adjustment

. analysis of irradied fuels.

The common approach to these three directions relies on the use of integral measurements to improve the uncertainty on GBG coming from nuclear physics.

III/1. Mock-up approach

This approach consists on measuring on one critical assembly typical of the considered power reactor the GBG at one time of the life of the plant, usually the beginning of life configuration (absorbers in) and to try to obtain bias factors.

For example, this approach was followed on ZPPR4, phase 1, for the CRBR reactor as part of the Demonstration Reactor Benchmark program. Assembly 4 corresponds roughly to a 300 MWe LMFBR : core height 91.4 cm, volume 2318 l. It was theoretically an EOL configuration but without fission products and Pu built-up in blankets. Detailed axial and radial distributions of the ^{238}U capture rates and ^{239}Pu fission rates were measured. Furthermore, at mid-plane , the radial distributions of $(1 + \alpha)$ were obtained for ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{235}U by the reactivity-reaction rate technique . After some calculated corrections, the breeding performances of this assembly can be "experimentally" obtained.

Finally the C/E discrepancy, expressed in term of breeding ratio, was :

$$C/E = 1.076 \pm 0.041 \ (1\sigma) \quad E = 1.172 \pm 0.045$$

calculations were performed using ENDF/BIII and three dimension diffusion code VENTURE. Of this 7.6 % difference, 5.4 % was due to an overpredition of the ratio ^{238}U capture to ^{239}Pu fission. Another 2.6 % came from a low calculation of the ^{238}U absorption to fission ratio, but here the experimental uncertainty is very much greater. Note that the uncertainty on ^{239}Pu absorption to ^{239}Pu fissions and $(1 + \alpha_9)$ appears to be slightly optimistic in this experiment. Furthermore a 0.8 % $\Delta K/K$ discrepancy remains in this C/E comparison. Note also that only small discrepancies appear on the integral reaction rate values : the main problems exist on local absolute value of reaction rates.

In contrast to ANL result, a SNEAK experiment shows only a 2% C/E discrepancy on GBG, but due to a 6 % overestimate of the core contribution and a 8 % underestimate of the blankets contributions.

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This is typically the problems of the mock-up approach : outside the difficulties to obtain a representative mock-up of a power reactor, this approach can lead to various C/E uncertainties only representative of a specific configuration. However, it is expected that the uncertainties in this approach may yet be reduced by improvements in both calculations and measurement techniques.

III/2. Parametric analysis of the neutron balance and data adjustment

This approach, followed for example in UK and FRANCE

for all the configurations of the life of the reactor, especially integral parameters related to GBG.

Global parameters of the neutron balance are measured systematically : material buckling and Koo parameters. Detailed parameters of the neutron balance are also measured in many cores configurations representative of all reactors power levels : fission rate ratios, capture rate to fission rate ratios for example.

Furthermore, all design parameters that contribute to the GBG value are also experimentally studied in a parametric approach · axial and radial reaction rate distributions in core and blankets, control rod influence on power distribution, higher Pu isotope problems, FP influence,...

All these integral experimental results are used to adjust evaluated nuclear data and defined bias factors and uncertainties on the GBG values calculated with these adjusted data. A typical example of this approach is represented by the CARNAVAL III system in operation at CEA to calculate the PHENIX and SUPER-PHENIX characteristics

This approach has the great advantage to can be directly applied to the whole range of power reactor levels.

III/3. Analysis of irradiated fuels

The measurements of the variation of fuel isotopic compositions versus burn-up have two main goals :

- accurate measurement of capture ratios,
- global check of the GBG calculations.

From the irradiation of pure isotopic samples and determination of the isotopic composition variation versus burn-up by chemical and mass spectrometry analyses, it is possible to get a really accurate measurement of the capture rate ratios for heavy isotopes in well defined spectrum conditions.

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This technique was previously used in fast irradiation reactor (EBR II, RAPSODIE, DFR).It is now used in demonstration reactor. Irradiations are planned in PFR. A lot of samples (45) irradiated in the central position of PHENIX (PROFIL experiment) give accurate results ($\pm$ 3 % to 2σ)., for all Pu, Uranium and some Americium isotopes in a representative neutron spectrum. These results confirmed the GBG calculation with the CARNAVAL version III system.

A systematic analysis of the standard fuel composition after irradiation allows a global determination of the GBG for core and blankets. This technique, applied on power plant as PHENIX, can give a final answer to the GBG, IBG and EBG uncertainties for this power level. From the systematic experimental program performed on PHENIX irradiated fuels during the first three years of operation,it was possible to obtain experimental values of external and internal breeding gains. The comparison of these values with the calculated ones (CARNAVAL III) confirms the validity of the parametric approach used and the breeding characteristics claimed for large power plants :

Phénix	Measurement	Calculation
IBG	- 0,420$\pm$ 0,035	- 0,41
EBG	0,565$\pm$ 0,02	0,54
GBG	0,145$\pm$ 0,04	0,13

The only problems in such an approach is the availability of various reactor plants with various spectra.

IV - ACCURACIES OBTAINED

According to the approach used in the various countries to improve the knowledge of GBG, the accuracies obtained can vary in a large amount.

However, when data adjustments are used, the present accuracies appear to vary between $\pm$ 0.03 (beginning of life) to $\pm$ 0.04 (end of life) for the whole range of power levels . These figures are confirmed on the results of Phenix irradiated fuel analyses.

According to the referred experiments, the mock-up approach results confirm or desagree with the previous figures :

ZPPR 4 C/E $\simeq$ 1.076 $\pm$ 0.041

SNEAK C/E $\simeq$ 1.02

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V - COMPARISONS OF BREEDING GAINS OBTAINED FOR VARIOUS LMFBR CONFIGURATIONS

The previous results concerned only mixed oxide two zones classical LMFBR.

Looking to other types of fuel, it appears from calculations that breeding ratio values up to about 1.5 for carbide fuels and nitride fuels with nitrogen fully enriched in 15N can be obtained for commercial reactors. For gas cooled fast breeder reactor, it is admitted that a $\simeq$ 1,4 breeding ratio can be obtained for commercial reactors. However all these reactors are currently regarded as back-up or long term systems compared to mixed oxide LMFBR.

Furthermore, using present mixed oxide fuel, a large improvement in the doubling time, the only parameter to consider for economic purposes, can be obtained using the heterogeneous core concept suggested by CEA . This concept uses internal breeder zones arranged to optimize power flattening and therefore minimize the core volume. Only one fissile enrichment zone is then needed. The build-up of plutonium in the internal breeder zones can largely reduce the reactivity loss per cycle.

The resultant reduction in the number of control rods further increased the external breeding gain axially. Typical doubling times lower than 15 years can be obtained in this way compared for example to 22 years for the standard two zoned concept. Furthermore, as a result of the high core enrichment in the fissile zones, the flux level is significantly decreased and this allows an increase in the burn-up rate to be obtained. Incidental additional advantage of the heterogeneous concept is that the sodium void coefficient is reduced by a large factor compared to the classical concept.

Retaining the familiar oxide fuel technology, this heterogeneous concept would allow to obtain the requested doubling time for LMFBR in the next future.