Latest Studies Related to the Use of Burnup Credit in France

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Abstract

Burnup Credit allows considering the reactivity decrease due to fuel irradiation in criticality studies for the nuclear fuel cycle. A working group composed of ANDRA, AREVA, CEA, EDF and IRSN studies the way to take fuel burnup into account in the criticality calculations considering fission products and an axial burnup profile. This paper proposes to give an outlook of the working group knowledge at the present state. It mainly concerns PWR UOX and MOX fuels, but Burnup Credit implementation for BWR UOX fuel is also investigated. As an application, this paper presents results of criticality calculations for storage configurations and transport casks obtained when applying the different conservatisms studied by the working group.

1. Introduction

In France, criticality safety analysis for nuclear fuel cycle facilities and transport casks usually consider fresh fuel. In some cases, however, limited to PWR UOX fuel, Burnup Credit is taken into account with some pessimistic hypothesis (only actinides are considered and the value of burnup used in the studies is equal to the mean burnup in the 50-least-irradiated centimetres) [1]. As the UOX fuel initial enrichment and the storage needs for spent fuel increase, operators have wished to develop a less penalizing method to implement Burnup Credit in criticality studies, by taking into consideration some fission products (the most neutron absorbent, stable or with a long half-life and not volatile) and a suitable bounding axial burnup profile.

In this context, a working group composed of ANDRA, AREVA, CEA, EDF and IRSN was formed to study the conservatism of all steps of the process to take fuel burnup into account in the criticality studies considering fission products and an axial burnup profile. These steps are: the definition of the axial profile of burnup in the studies, the depletion calculations, the criticality calculations. This paper proposes to give an outlook of the present state of knowledge of the working group.

The discussions of the working group have first focused on PWR UOX fuels. In addition, studies on Burnup Credit for PWR MOX fuels are now under way. Moreover, in the framework of optimization in some parts of the fuel cycle, namely reprocessing plants, Burnup Credit implementation for BWR UOX fuels is being investigated.

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2. PWR UOX fuel studies

The early studies are presented in reference [2]. They mainly concern the definition of the steps of the process to take Burnup Credit into account in a criticality calculation with CRISTAL V1 package [3] as illustrated in Figure 1. The CESAR or DARWIN depletion codes calculates the concentrations of isotopes at the end of irradiation or after a cooling time, then these concentrations are used as input data in the CRISTAL criticality package which provides the neutron multiplication factor associated to the calculated situation.

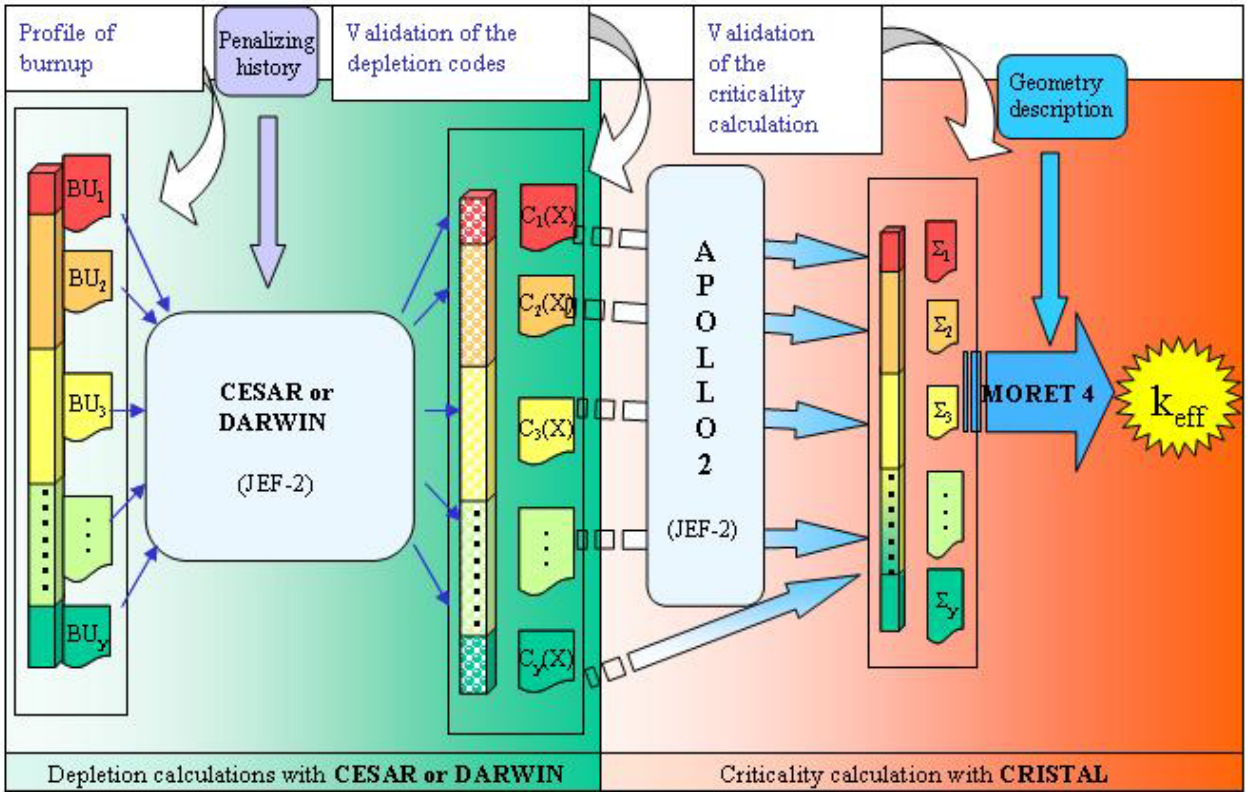

Figure 1 : Steps of the process to take Burnup Credit into account in a criticality calculation with CRISTAL

The early studies also concern the influence of the irradiation conditions on the conservatism of the depletion calculations. It was decided to quantify the effects considering the spectrum hardening, the fuel temperature variations (Doppler effect), the temporal variation of the flux level (variations of specific power) and the cooling time. The conclusion is that it is conservative to consider for the depletion calculations: the presence of the control rods, a maximized boron concentration, moderator temperature, fuel temperature and specific power (when no cooling time is considered) and the minimum of the cooling time that can be justified by the operators (for devices which are to be used for cooling times of less than 50 years).

The latest studies mainly concern: the determination of an axial profile of burnup for the criticality studies and the determination of correction factors for the isotopic composition in Burnup Credit applications.

2.1 Determination of an axial profile of burnup

The definition of the axial profile of burnup is one of the major criticality safety issues for Burnup Credit applications. The feasibility of the determination of an axial profile of burnup by using a French database of axial burnup measurements has been studied.

The experimental database, resulting from teamwork between EDF R&D and AREVA-NC [4], includes axial burnup measurements of hundreds of assemblies irradiated with in 900 and 1300 MWe PWR reactors (EDF reactor fleet).

First investigation [5] has concerned assemblies with an initial ^{235}U enrichment of 3.1% and an average burnup ranging between 30 and 40 GWd/t.
In order to evaluate the statistic variation and the "validity" of the experimental average burnup profile, two criteria have been used: a global criterion, named Average Standard Deviation, and a local criterion, named Local Relative Deviation (see [5] for the definition of these criteria). Even if one third of these assemblies may have been exposed to control rods during the last cycle of irradiation and 10% may have been exposed to control rods during more than two thirds of the time spent under irradiation, the database analysis has showed that the shapes of all profiles of burnup are very similar.
Then, using this database, two profiles have been deduced (Figure 2). If we compare these profiles with the OECD profiles of burnup currently used in the benchmark criticality studies, the main characteristic of these profiles of burnup is to be uniformly burnt on the entire height of the bundle. These profiles of burnup deduced from the database have been tested in the case of a standard criticality environment in order to evaluate the end effect and to compare the computed profile.

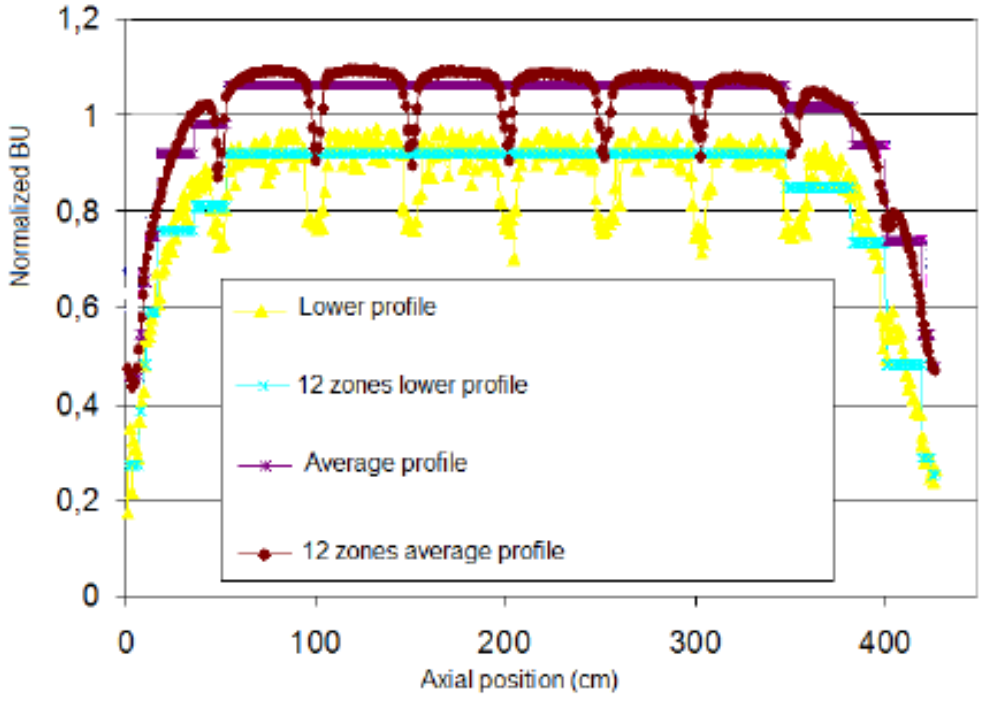

Figure 2 : Average and lower profiles deduced from an experimental database

Recently, the approach has been extended to the other families of enrichment and burnup.
In addition to the previously exposed criteria of selection, a search for profiles deviating too much from the average profile has been implemented.

Moreover, because the lower profile deduced from the database is not a real profile (it has not been measured), new criteria have to be evaluated in order to identify a penalizing profile. These criteria are:

- the consideration of the minimal burnup value in the last 50 centimetres of the assembly (integral criterion),
- a modelling of the extremities of the profile by lines determined by root mean square method (shape criterion).

These criteria have to be crossed to identify the most penalizing profile.
Finally, the gain between a flat profile and the most penalizing profile with a Burnup Credit complete study calculation has to be estimated.
These studies are currently under analysis, but some results are already available for PWR 900 MWe (limited to an initial enrichment of 3.2%) [6].

2.2 Determination of isotopic correction factors

In order to ensure the conservatism of the inventory of the irradiated fuel, the depletion calculations have to be validated with isotopic composition data measured by Post Irradiation Examination (PIE). The validation of depletion codes relies on comparisons between calculated and measured values of the concentrations. For each sample of irradiated fuel, a relative error between calculations and measures, called (C-E)/E is calculated. This validation can be used to determine isotopic correction factors that should be applied, if needed, to the calculated concentration of each actinides and fission products considered. Two sets of correction factors have been determined.

2.2.1 Determination taking into account the depletion codes validation

The conservative method presented in [7], uses the CEA experimental database of the DARWIN 2.0 package to estimate the required isotopic correction factors for the depletion calculations. This evaluation is based on statistical analysis of computational biases (C-E)/E as a function of burnup for a given isotope. It takes into account the total uncertainties on the (C-E)/E values that correspond to the combination of uncertainties on chemical assays, determination of the burnup and calculations used in the validation process.

The kernel-smoothing model seems to be the best way to determine accurate bounding isotopic correction factors. In order to maximize the k_{eff} value, fissile isotopes have to be corrected by a factor greater than 1, whereas absorbing isotopes have to be corrected by a factor lower than 1. Consequently, overestimated fissile isotopes and underestimated absorbing isotopes will not be corrected (factor equal to 1). The penalty on the Burnup Credit gain of this isotopic correction is less than 2.8%. Note that this penalty could be reduced with a reassessment of the uncertainties of the PIE data and/or achievement of new PIE data.

2.2.2 Determination taking into account the depletion and the criticality codes validation

The fuel inventory, calculated with the DARWIN package, is used as an input data for the criticality-safety package CRISTAL, which provides the neutron multiplication factor associated to the calculated situation. The determination of correction factors for UOX fuel in [8] is based on the experimental validation of the spent fuel inventory carried out using the DARWIN package, and on the experimental validation of the reactivity worth carried out with the CRISTAL package. Recently, these correction factors have been re-determined from the experimental database using the JEFF3.1.1 library [9].

3. PWR MOX fuel studies

To date, the studies on PWR MOX fuels concern the determination of a conservative inventory of the irradiated fuel. The conservatism of the inventory of the irradiated fuel depends on the irradiation conditions and the fresh fuel characteristics. Both aspects have been investigated by the working group.

3.1 Influence of irradiation conditions

Similarly to PWR UOX fuel, the influence on the conservatism of depletion calculations of the main parameters which are responsible for changes in the fuel inventory (control rods insertion, moderator temperature, boron concentration, fuel temperature, specific power) is studied for a given MOX fuel composition [10].

The presence of control rods during the irradiation of a MOX assembly induces the capture of thermal neutrons and thus the hardening of the neutron spectrum. The increase of the moderator temperature reduces the moderating power of water (due to the decrease of the density) that also induces the hardening of the neutron spectrum. In the same way, as the presence of control rods, the increase of the boron concentration induces the capture of thermal neutrons and consequently the hardening of the neutron spectrum. The spectrum hardening leads to increase the fissile plutonium content and consequently to significantly increase the reactivity.
The increase of the fuel temperature induces a broadening of the ^{238}U capture resonance that leads to increase the neutron capture and thus the forming of fissile nuclides following the chain reaction.
Increasing the specific power is like decreasing the irradiation time. Since the decrease of the irradiation time reduces the forming by decay of some absorbent actinides or fission products, the reactivity of the burned fuel increases.

Finally, conditions that maximize the reactivity of the burned fuel are: the presence of control rods during the irradiation, a maximized moderator temperature, boron concentration, fuel temperature and specific power.

The penalty due to control rods insertion accounts for 65% of the total penalty. The one due to the moderator temperature accounts for 20% and the one due to boron concentration for 10%. This distribution of penalties looks like the one obtained in the case of PWR UOX fuel.

3.2 Methods to determine a bounding isotopic MOX composition

One of the main difficulties of a Burnup Credit implementation for PWR MOX fuel is the wide range of parameters compared to PWR UOX fuel (initial plutonium composition, plutonium content, uranium composition, presence of ^{241}Am, zoning of assemblies). Contrary to PWR UOX fuel, a fresh fuel with a conservative isotopic vector (of plutonium) will not inevitably lead to the most reactive fuel after irradiation for PWR MOX fuels: the fissile isotopes of plutonium will disappear and give absorbent isotopes, while captures on absorbent plutonium isotopes will gives new fissile isotopes… Today, to use MOX burnup in criticality studies, fuel depletion should be calculated for each initial MOX fuel composition. Due to the wide variety of MOX fuel compositions, this solution seems unachievable without huge calculation efforts. Therefore, to simplify MOX Burnup Credit implementation, a solution is the definition of a bounding fresh MOX fuel that remains conservative during depletion. Two methods to determine a bounding plutonium vector for the fresh fuel that gives after irradiation the most reactive fuel whatever the irradiation history have been developed.

3.2.1 Method based on a reactivity equivalence between an UOX enrichment and a MOX fuel composition

The method [11] consists in giving a weight in terms of reactivity to each MOX fuel isotope by determining a reactivity equivalence between an UOX enrichment and a MOX fuel composition, for a given configuration. Indeed, for a given UOX population, criticality studies are performed considering the boundary composition which is obtained for the higher initial enrichment for both fresh and irradiated level (whatever the irradiation level is).

The proposed method consists in three steps.

First of all, for each isotope present in the MOX composition, a weight in terms of reactivity must be set in order to construct a reactivity equivalent equation between a MOX composition and an UOX composition. So, for each MOX composition, an equivalent uranium enrichment (%($^{235}U_{eq}$)) is determined. In this aim, for each proportion X of isotope present in the MOX fuel composition, a coefficient α is defined. They represent a relative weight in term of reactivity for each isotope. Then, all the contributions (αX) on the reactivity are summed up. The different isotopes considered are: ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu and ^{241}Am. Correlations are considered between isotopes and the order of the correlations can be two (αX_1X_2), three ($\alpha X_1X_2X_3$) or more to improve the accuracy of the method. Finally, the equation used is expressed as follows:

$$\%\,(\,^{235}U_{eq}) = \sum_i \alpha_i X_i + \sum_{jk} \alpha_{jk} X_j X_k + \sum_{jkl} \alpha_{jkl} X_j X_k X_l + \ldots$$

with:

X: proportion (in %wt) of each isotope in the MOX fuel composition (uranium, plutonium or americium),

α: weight of each isotope in the global reactivity of the MOX fuel compositions.

Secondly, MOX fuel irradiation is taken into account. A correlation equation corresponding with the decay of the reactivity versus the irradiation of each MOX composition is necessary. In order to reduce calculations, MOX compositions used to establish this equation are the same as for the first equation. To define this second equation it is important to have a look on the decay of the reactivity versus the irradiation. Indeed, the aim is to directly access the uranium enrichment equivalent %($^{235}U_{eq}$)$_{BU=I}$ for a given irradiation level (referred as "I"), without making all the depleted calculations. As previously, for each isotope, coefficients β are determined. They represent the relative (between fresh fuel, BU=0, and fuel irradiated up to "I") depleted reactivity of each isotope (X) present in the MOX composition (^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am). Correlations between isotopes can be taken into account too. The order of the correlations can be two (βX_1X_2), three ($\beta X_1X_2X_3$), or more too. Again, the different contributions on the reactivity of each isotope and the correlations are summed up. Finally, the second equation used is expressed as follows:

$$\left|\frac{\%\,(\,^{235}U_{eq})_{BU=I} - \%\,(\,^{235}U_{eq})_{BU=0}}{\%\,(\,^{235}U_{eq})_{BU=0}}\right| = \sum_i \beta_i X_i + \sum_{jk} \beta_{jk} X_j X_k + \ldots$$

with:

X: proportion (in %) of each isotope in the MOX fuel composition (uranium, plutonium or americium)

β: weight of each isotope in the global decreasing reactivity of the MOX fuel compositions.

Then, this methodology is applied for each MOX fuel composition and the most reactive composition is finally selected.

These equations have been applied on realistic MOX isotopic compositions [11] in order to estimate the validity of the method and its area of application.

3.2.2 Method based on plutonium isotopes relative abundance

Only three isotopes (^{239}Pu, ^{240}Pu and ^{241}Pu) were initially taken into account for the fresh bounding plutonium vector. The work done [12] shows that for commonly used plutonium (plutonium coming from the reprocessing of commercial UOX fuel assemblies) and cooling times limited to 100 years:

- the configuration leading to the maximum neutron multiplication is always obtained with a minimum amount of ^{240}Pu,
- the determination of bounding initial isotopic vectors of the plutonium depends on the spent fuel cooling time mainly due to the beta-decay of the ^{241}Pu (whose half life is 14.4 years) in ^{241}Am, a neutronic absorber for thermal spectrum.

Further studies [10] have checked that the presence of ^{238}Pu and ^{242}Pu in the fresh MOX fuel doesn't impact on the above conclusions.

Thus, a simple way to define a bounding plutonium vector for a fresh fuel with a maximum value of Pu/(U+Pu), a minimum burnup and a minimum cooling time could be:

- select the minimum amount of ^{240}Pu ($r_{0\ min}$),
- determine the minimum and the maximum value of $r_1 = {}^{241}Pu/{}^{240}Pu$,
- perform two depletion and criticality calculations with the isotopic vectors V1 and V2 described in Tableau 1.

Tableau 1 : Bounding plutonium vector based on plutonium isotopes relative abundance

V1	V2
$^{240}Pu/Pu_{tot} = r_{0\ min}$	$^{240}Pu/Pu_{tot} = r_{0\ min}$
$^{241}Pu/Pu_{tot} = r_{0\ min} \times \mathbf{r_{1\ min}}$	$^{241}Pu/Pu_{tot} = r_{0\ min} \times \mathbf{r_{1\ max}}$
$^{242}Pu/Pu_{tot} = 0$	$^{242}Pu/Pu_{tot} = 0$
$\mathbf{^{239}Pu/Pu_{tot} = 100 - r_{0\ min} \times (1 + r_{1\ min})}$	$\mathbf{^{239}Pu/Pu_{tot} = 100 - r_{0\ min} \times (1 + r_{1\ max})}$

For long cooling times, the bounding initial plutonium vector would minimize the amount of ^{241}Pu, and for short cooling times, the bounding vector would minimize or maximize the amount of ^{241}Pu depending on both the burnup value and the weight ratio Pu/(U+Pu) as illustrated in Figure 3.

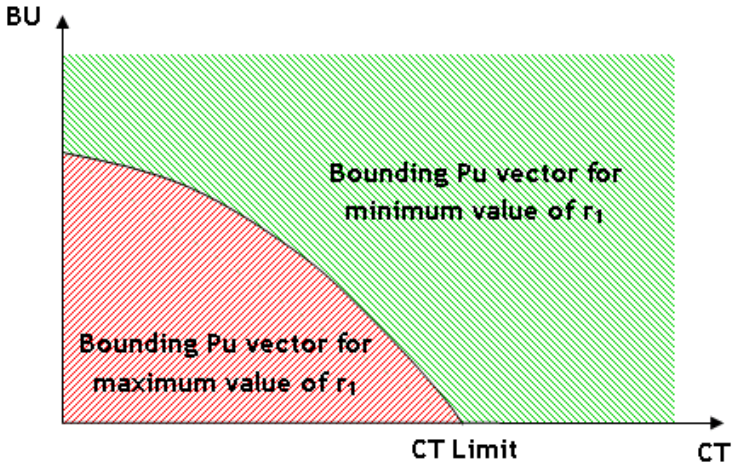

Figure 3 : Bounding plutonium vector zones (burnup (BU); cooling time (CT)) obtained for the maximum or the minimum value of $r_1 = {}^{241}Pu/{}^{240}Pu$

Calculations performed for different irradiation conditions on a sample of MOX fuel compositions representative of MOX used in French reactors [10] has allowed to verify the conservatism of the method and to evaluate the associated penalty. This penalty decreases with burnup and cooling time.

4. BWR UOX fuel studies

The major difference in the operation of a BWR compared to a PWR is the steam void formation in the core. The latest is characterized by the steam to the total water mass ratio, called void fraction, which could varies from 0% to about 70% between the bottom and the top of the core. This results in a highly heterogeneous and time varying reactor configuration that makes fuel depletion analysis in BWRs more complicated than in PWRs.

Thus, in addition the analysis of the validity of the assumptions made to define the axial profile of the burnup, to determine the composition of the irradiated fuel and to compute the criticality simulation, BWR Burnup Credit implementation requires to carefully review the validity of the assumption made to define the axial profile of void fraction. Part of these items has been investigated.

Since the few studies on Burnup Credit issues in BWR fuel [13] show that the predominant parameter is the density of the moderator linked to the void fraction, and that other parameters like the fuel temperature, the specific power and the irradiation history are secondary, the studies have focused on the void fraction parameter.

The studies [14] based on a General Electric 8×8 assembly have allowed to:

- study the influence of the void fraction during the irradiation on the spent fuel content in order to get a conservative estimate of this content,
- compare the results obtained with two different depletion codes (TRITON t-depl sequence from SCALE package allowing two dimensional calculations and VESTA, a generic Monte-Carlo depletion interface code developed at IRSN allowing up to three dimensional calculations) with a view to determine the more suitable calculation method for Burnup Credit implementation for BWR fuel.

The main result from these studies [ii] is that the assumption of the maximum void fraction in the depletion code does not always provide a bounding value of k_{eff} of an irradiated fuel assembly model depending on the profile of burnup considered. In addition, a good agreement is observed between TRITON and VESTA results. Finally, a three-dimensional calculation with VESTA shows that, for this case, the different axial regions of void fraction don't have much influence between them.

This study already highlights the difficulty to find a generic approach to implement Burnup Credit for BWR, even with a standard design of assembly.

5. Burnup Credit gain estimation

The Burnup Credit gain is defined as the difference of reactivity between an irradiated fuel and the same fuel fresh (Δk).

The values of Δk for different configurations of storage (storage pool configuration with a borated basket containing 9 fuel assemblies) and transportation (transport cask with UOX or MOX assemblies) are given in Tableau 2 for a PWR UOX fuel initially enriched at 4.5% and for a PWR MOX fuel with a value of Pu/(U+Pu) equal to 4.5%. For those calculations, the

[ii] The results are obtained for a specific BWR assembly. Therefore, they are not applicable to any design of BWR assembly.

pessimistic assumptions regarding the conditions of irradiation described in introduction of paragraph 2 and in paragraph 3.1 [iii] were taken into account and a standard PWR profile with 17 axial zones was used.

Tableau 2 : Burnup Credit gain estimations for industrial applications (burnup equals to 44 GWd/t, no cooling time)

Configuration	Δk (pcm)	
	UOX	MOX
Assembly surrounded by 20 cm of water	18300	10000
Storage pool (fallen assembly)	19200	10400
Storage pool (off-centered assemblies)	18300	10200
Storage pool (higher standing assemblies)	16000	8800
Transport cask	19600	10900

Even if the conditions of irradiation are penalizing and no realistic, the Burnup Credit gain estimations for industrial applications are significant. One can notice though that gains are almost twice as much important for UOX fuel than for MOX fuel. A study of the origin of this Burnup Credit gain shows that the Burnup Credit gain mainly comes from actinides in the case of UOX fuel (around 65%) and from fission products in the case of MOX fuel (around 65%).

As for BWR fuel, the values of Δk have not been computed for industrial application, but for an infinite array of assemblies. For that configuration, they reach roughly 14000 pcm if a standard profile of burnup and void fraction are taken into account. This gain is reduced of about 500 pcm only if the assumption of the maximum void fraction is made.

6. Conclusion and prospects

The present state of knowledge of the French working group on Burnup Credit covers all steps of the process to take fuel burnup into account in the criticality studies considering fission products and an axial burnup profile as far as PWR UOX fuel is concerned. The objective is now to make use of this knowledge by operators to implement Burnup Credit in criticality studies.

As far as PWR MOX fuel is concerned, the knowledge of the working group only covers the determination of a conservative inventory of the irradiated fuel. The methods to determine a bounding isotopic MOX composition have to be evaluated in order to check if the Burnup Credit gain available is enough for MOX industrial applications. For that purpose, a comparison study is underway. Both methods will be applied on realistic MOX fuel compositions and realistic configuration from the nuclear fuel cycle installations.

Finally, concerning BWR fuel, the knowledge of the working group is limited to preliminary studies of the influence of the burnup and the void fraction (during irradiation) on BWR spent fuel content, on the effective multiplication factor of an infinite array of BWR assemblies and a code-to-code comparison for BWR fuel depletion calculations relevant to Burnup Credit. Further studies of the working group on this topic should focus on the improvement of the comprehension of the combined influences of an axial profile of the burnup and void fraction by studying other profiles, the examination of other BWR assembly designs with more heterogeneities (depending on the available data) and the investigation of the influence of other parameters like control blades insertion, neighbourhood assemblies as well as the irradiation history.

[iii] Moderator temperature equal to 330 °C, boron concentration equal to 1200 ppm and control rods in.

References

[1] L. Maubert, "The Burn-Up consideration in the criticality safety of irradiated LWR fuel cycle plants", Proceedings of International Seminar on nuclear Criticality Safety, ISCS87, Tokyo, Japan, October 1987.

[2] J. Raby et al., "Current studies related to the use of Burnup Credit in France", Proceedings of International Conference on Nuclear Criticality Safety, ICNC2003, Tokai Mura, Japan, October 2003.

[3] J.-M. Gomit et al., "CRISTAL V1: Criticality Package for Burn up Credit calculations", Proceedings of International Conference on Nuclear Criticality Safety, ICNC2003, Tokai Mura, Japan, October 2003.

[4] H. Toubon et al., "Burn-up measurements coupling gamma spectrometry and neutron measurement", European Nuclear Conference, Versailles, France, December 2005.

[5] E. Cabrol et al., "Determining an axial burn-up profile for BUC criticality studies by using french database of axial burn-up measurements", Proceedings of International Conference on Nuclear Criticality Safety, ICNC2007, Saint Petersburg, Russia, May 2007.

[6] C. Riffard et al., "Recommended bounding axial profiles in BUC applications from actual burnup measurement of French PWR assemblies", Proceedings of International Workshop on Advances in Applications of Burnup Credit for Spent Fuel Storage, Transport, Reprocessing and Disposition (CSN / IAEA), Cordoba, Spain, October 2009.

[7] J. Raby et al., "Determination of correction factors for the isotopic composition in Burnup Credit applications taking into account the depletion codes validation", Proceedings of International Conference on Nuclear Criticality Safety, ICNC2007, Saint Petersburg, Russia, May 2007.

[8] A. Barreau et al., "PWR-UOX fuel validation of the BUC calculation route "CIRACUSE" with MINERVE and REBUS experiments", Proceedings of International Conference on Nuclear Criticality Safety, ICNC2007, Saint Petersburg, Russia, May 2007.

[9] C. Riffard et al., "Correction Factors derived from French experiments with recent JEFF3.1.1 library for PWR-UOx BUC applications", Proceedings of International Workshop on Advances in Applications of Burnup Credit for Spent Fuel Storage, Transport, Reprocessing and Disposition (CSN / IAEA), Cordoba, Spain, October 2009.

[10] L. Jutier et al., "Conservative approach for PWR MOX Burnup Credit implementation" Proceedings of International Conference on the Physics of Reactors, PHYSOR2008, Interlaken, Switzerland, September 2008.

[11] F. Gevaert et al., "Methodology to determine a boundary isotopic MOX composition for a given MOX irradiated fuel population", Proceedings of International Conference on Nuclear Criticality Safety, ICNC2007, Saint Petersburg, Russia, May 2007.

[12] J. Raby et al., "Bounding plutonium vectors for PWR-MOX Burnup Credit applications", Proceedings of International Conference on Nuclear Criticality Safety, ICNC2007, Saint Petersburg, Russia, May 2007.

[13] J.C. Wagner et al., "Investigation of Burnup Credit Modeling Issues Associated with BWR fuel", ORNL/TM-1999/193, 2000.

[14] L. Jutier et al., "Investigation of Burnup Credit implementation for BWR UOX fuel", Proceedings of Nuclear Criticality Safety Division Topical Meeting, NCSD2009, Richland, Washington, USA, September 2009.