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Current Studies Related to the Use of Burnup Credit in France

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Abstract

In order to avoid criticality risks, a large number of facilities using spent fuels have been designed considering the fuel as fresh. This choice has obviously led to considerable safety margins.

In the early 80's, a method was accepted by the French Safety Authorities allowing to consider the changes in the fuel composition during the depletion with some very pessimistic hypothesis: only actinides were considered and the amount of burnup used in the studies was equal to the mean burnup in the 50-least-irradiated centimeters.

As many facilities still want to optimize their processes (e.g. transportation, storage, fuel reprocessing), the main companies involved in the French nuclear industry, researchers and IRSN set up a Working Group in order to study the way burnup could be taken into account in the criticality calculations, considering some fission products and a more realistic axial profile of burnup.

The first part of this article introduces the current French method used to take burnup into account in the criticality studies. The second part is devoted to the studies achieved by the Working Group to improve this method, especially concerning the consideration of the neutron absorption of some fission products and of an axial profile of burnup: for that purpose, some results are presented related to the steps of the process like the depletion calculations, the definition of an axial profile and the criticality calculation. In the third part, some results (keff) obtained with fission products and an axial profile are compared to those obtained with the current one.

The conclusions presented are related to the present state of knowledge and may differ from the final conclusions of the Working Group.

KEYWORDS: Burnup Credit, Axial profile, Fission products, French Working Group.

1. Introduction

Up to the 80's, the nuclear facilities which were dealing with spent fuel were designed with the assumption of fresh fuel. This assumption led to considerable safety margins.

In the early 80's, in order to use the existing devices at La Hague reprocessing plant for some irradiated UO_2 fuel initially enriched at 4.4 % (this enrichment was higher than the highest enrichment of 3.1 % considered at the designing stage), a method was proposed by COGEMA to enable them to consider a certain amount of burnup in the criticality studies. However some pessimistic assumptions were made to guarantee some safety margins:

- only uranium and plutonium were considered after the depletion of ^{235}U and ^{238}U during the irradiation,
- the amount of burnup used in the criticality studies was lower than the value reached in the 50-least-irradiated-centimeters,
- the value of the mean burnup in the 50-least irradiated-centimeters was verified by a measurement.

The calculations were supported by the HTC experiments which were achieved in Apparatus B in Valduc. Those subcritical experiments[1)] involved fuel pins, representative of a fuel initially enriched at 4.5 % and irradiated at 37.5 GWd/t, manufactured only with actinides. The pins arrays were arranged in different types of configurations which were representative of reprocessing, storage and transport.

This actinide-only method was accepted by the

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French Safety Authorities and was, afterwards, used for the transport of irradiated fuel and, also, at the designing stage of the UP3 and UP2-800 at La Hague reprocessing plant.

But, since the initial enrichment is increasing and the needs of interim storage of irradiated fuel is growing, it is becoming necessary for the nuclear industry to reduce the conservatisms due to the very pessimistic hypotheses of this actinide-only method.

For that purpose, a Working Group was created in 1997, gathering most of the French nuclear companies, to analyze different propositions of introduction of some fission products plus a more realistic axial profile of burnup in criticality studies.

2. Work in Progress

The new points introduced by the method are related to:

- the neutron absorption of some fission products ; the method proposes to first take into account the 6 following fission products ^{149}Sm, ^{152}Sm, ^{103}Rh, ^{143}Nd, ^{133}Cs and ^{155}Gd, which are responsible for 50 % of the absorption of all fission products. This list can be extended to the 9 other fission products which are considered by the OECD BUC W. G. (^{99}Tc, ^{145}Nd, ^{153}Eu, ^{95}Mo, ^{147}Sm, ^{150}Sm, ^{151}Sm, ^{109}Ag and ^{101}Ru),
- a more realistic description of the axial profile of the burnup.

As those new considerations will reduce the conservatisms, it seems necessary to guaranty the validity of the assumptions made to:

- determine the composition of the irradiated fuel (especially for fission products),
- define the axial profile of the burnup,
- and compute the simulation (particularly regarding the knowledge of the cross-sections of the isotopes that are being taken into account).

For that purpose, the following paragraphs describe all the assumptions made during (i) the depletion calculations (ii) the definition of the axial profile of burnup in the studies, (iii) the criticality calculations; those assumptions have to ensure a global conservatism of the method. Moreover, the amount of burnup used in the method will have to be guaranteed (e.g. by some measurements). Thus the safety of the method will be assured.

The objective of the French Working Group is to study all steps of the process to take fuel burnup into account in the calculations, the current studies are related to UO_2 PWR fuel.[2)]

2.1 Depletion calculations step

First of all, it seems necessary to guaranty in the process that, for a given amount of burnup, the depletion calculations lead to a conservative inventory of the irradiated fuel.

It can be noticed that, in the depletion calculations, the burnup is determined by the calculated ratio of some isotopes concentrations (e.g. Nd); thus the qualification of the method will be tightly linked to the chosen "indicator of burnup".

The conservatism of the depletion calculations depends on:

- the conditions of irradiation (internal EDF studies); the Group decided to quantify there effect considering (i) the spectrum hardening, (ii) the fuel temperature variations (Doppler effect), (iii) the temporal variation of the flux level (variations of specific power), (iv) the cooling time,
- the qualification of the depletion codes which are used for the calculations i.e. CESAR [3)] (industrial code) or DARWIN [4)] (reference code).

2.1.1 Irradiation history

The main parameters which are responsible for changes in the fuel inventory (neutron spectrum, fuel temperature, specific power, cooling time) are discussed in this paragraph.

Spectrum effect

It has already been shown [1)] that, for a given burnup, the fuel is more reactive when it is irradiated with a hardened neutron spectrum. Consequently, the different conditions of irradiation that might harden the spectrum have been sought.

During the irradiation, the spectrum may be hardened when:

- control rods are inserted; this situation leads (for UO_2 initially enriched at 4.5 % and irradiated at 44 GWd/t) to a penalty up to Δk = 5 % [2)], others evaluations (with others configurations) [5)] showed, that if control rods are inserted during 1 cycle the penalty becomes Δk = 0.5 % (first cycle) to Δk = 2.1 % (third cycle).
- an UO_2 assembly has been irradiated near some MOX assemblies; the penalty with a very pessimistic assumption (the UO_2 fuel is surrounded by 8 MOX assemblies during the entire irradiation) leads to a Δk of 1.2 % [5)] for an infinite array of UOX assemblies irradiated up 40 GWj/t (others studies [2)] give the same tendency when MOX assemblies were present on the 4 sides of the UO_2 fuel during 4 cycles),
- the temperature of the water (moderator) is increased (the temperature is set to its maximum value at the edge of the reactor core),
- the concentration of the boron during the cycle is increased,
- burnable poisons are present.

The Working Group considers as conservative, for the depletion calculations, the presence of the control rods, the maximum concentration of the boron, a temperature of the water sets to its out-of-core value.

Doppler effect

The temperature of the fuel used during the depletion calculations has a slight effect on the reactivity. Actually, it is conservative to consider a high value of the temperature as it leads to more

absorptions on the ^{238}U (and then to more productions of ^{239}Pu). Consequently, the temperature of the fuel may be set to the maximum value of the effective temperature of the fuel.

Temporal effect

The temporal effects are linked to the variation of the specific power (SP). Actually, this effect is the consequence of the competition between neutron absorptions (which are directly related to the value of the burnup) and decay reactions (related to the time spent in the reactor).

Some studies [2] showed that an increase of the specific power is conservative (when no cooling time is considered) even if fission products are taken into account. The results obtained for a single assembly surrounded by 20 cm of water (initially enriched at 4.5 % and irradiated at 44 GWd/t) are given below.

Table 1 Single assembly (depleted with the control rods inserted - 15 Fissions products)

	keff ($\sigma < 0.00095$)		
	SP ↓	11 GWd/t	33 GWd/t
cooling time: 0 year	5 W/g	0.8639	0.7757
	20 W/g	0.8648	0.7837
	40 W/g	0.8664	0.7832
	60 W/g	0.8668	0.7820
cooling time: 5 years	5 W/g	0.8626	0.7603
	20 W/g	0.8623	0.7637
	40 W/g	0.8589	0.7634
	60 W/g	0.8587	0.7595

It emerges from the table that, for a cooling time of 5 years, the highest specific power does not give the most reactive result. This has already been shown in some previous publications [6] and can be explained by the behavior of some isotope concentrations when the specific power varies.

While considering the isotope concentration variations, the calculations pointed out that, when the specific power increases, the concentrations of the absorbent isotopes decrease and the fissile ones increase, except for the following isotopes:

- ^{149}Sm and ^{152}Sm which increase respectively by 52 % and 5.5 % when the specific power grows from 5 W/g to 60 W/g (at a burnup of 40 GWd/t),
- ^{242}Pu which slightly increases by 2.8 % when the specific power grows from 5W/g to 40 W/g (at a burnup of 45 GWd/t),
- ^{155}Eu (father of ^{155}Gd), which increases by 26 % (at 20 GWd/t) and 44 % (at 44 GWd/t) when the specific power grows from 5 to 60 W/g.

The different behaviors observed in the table have been published and analyzed in.[7]

Then the assumption of a maximized specific power, which leads, for no cooling time, to a conservative value of the keff is no more conservative when a 5 year cooling time is considered.

Cooling time

Many studies have shown [8] that, after the irradiation and up to a cooling time of 100 years, the reactivity decreases. This decrease is mainly due to the decay of the ^{241}Pu (to ^{241}Am) plus the increase of ^{155}Gd.

After 100 years of cooling time, the reactivity starts to increase again (as the ^{241}Am decays) until around 30 000 years.

The needs, in terms of cooling time, for the French Working Group participants mainly concern some devices which are to be used for cooling times of less than 50 years. For those applications, it should be acceptable to consider, in the criticality studies, the minimum of the cooling time that can be justified by the operators[1].

However, for interim storage, this approach raises the problem of how operators can guarantee that, after 100 years of cooling time, a solution will exist to move the fuel to some other adapted storage? For this type of storage and for cooling times greater than 100 years, some special studies will have to be done to determine the maximum of the reactivity when the cooling time is growing. It should be decided not to take into account the ^{241}Am in those studies.

2.1.2 Qualification of the Depletion Codes

The depletion codes that have been used yet are CESAR [3] or DARWIN [4] Codes. They have been qualified on an experimental basis.[9] This qualification rests on comparisons between calculated values of the concentrations and measured ones.[10] Those comparisons have been achieved, on both (i) punctual analysis of irradiated fuel (with initial enrichment of 4.5 % and burnup up to 60 GWd/t) (ii) global analysis during the dissolution of irradiated assemblies (with initial enrichment comprised between 3.1 % and 3.5 % and burnup up to 45 GWd/t).

These comparisons will be used to determine correction factors for the calculated concentrations of each of the actinides and fission products which are considered in the method.

2.2 Definition of an axial profile of burnup

Until now the value of the burnup, which was applied on the whole length of the assembly, was equal to the mean value of the 50-least-irradiated-centimeters of the assembly.

This assumption was very conservative for an actinide only methodology and the geometrical configurations studied, giving a reduction of 22 % of the mean burnup for a standard profile (e.g. 34 GWd/t will be applied to the whole length of the assembly whereas the real mean burnup would be equal to 44 GWd/t).

In order to consider a more realistic profile, it was

[1] For no cooling time, the amount of ^{239}Pu has to be increased with the amount of ^{239}U and ^{239}Np.

decided to determine a profile for the criticality studies by:

- examining the profiles measured at La Hague reprocessing plant (more than 3000 irradiated assemblies have been measured),
- calculating different types of profiles to determine a penalizing one.

Those two studies pointed out that:

- the systematic use of a penalizing profile is very pessimistic (the value of the keff can raise of 12 % in Δk) [11] and it is needed to ensure that the profile will always be penalizing for every possible reactor management,
- As part of burnup axial profile study, 3000 profiles have been measured at La Hague - COGEMA plant. These measurements pointed out that most of the profiles were quite similar. The statistical study of these profiles will allow us to determine a conservative axial profile (for "most of the profiles" already measured). The conservatism of the axial profile used in the studies will have to be demonstrated (e.g. checked by a measure of each assembly and the measured burnup will have to be greater (at different points along the fuel assembly) than the one used in the criticality studies). The number of zones used for the profile model will be determined to give a value of the keff which is not too conservative : if important margin exists for the configuration studied, there is no need to consider a high number of zones (the keff will decrease as N increases).

2.3 Criticality calculation step

The new French CRISTAL V.1 package will allow to perform automated criticality calculations using burnup credit like in figure 1. This package uses the depletion code CESAR 5 or the DARWIN 2.0 system, the APOLLO 2 [12] computer code for cross sections calculations from 172 energy-groups libraries, and the multigroup Monte Carlo computer code MORET4.[13]

Next paragraphs will present the steps of the process described in figure 1, the fission product programs for qualification and the problem of loosely coupled units.

2.3.1 Calculation process

CRISTAL V.1 [14] contains an interface [15], coupled with the depletion code. It is designed to issue automatically data files for the criticality calculations. It will allow choosing and automatically taking into account:

- correction factors (applied to the concentrations of the actinides and the fission products chosen) determined by the qualification of the evolution calculations and by the qualification of the cross-sections,
- the axial profile (flat profile, penalizing profile without measurement verification, conservative profile with precise measurement guarantied [16]),
- the number of axial zones chosen.

2.3.2 Qualification of the Fission Products Cross Sections

The fission product programs [17] are based on two types of experimental data.

IRSN carries out experiments in Valduc [18, 19] using different fission product isotopes to validate the calculation scheme. The criticality calculation chain has already been qualified in 1991 with ^{149}Sm.

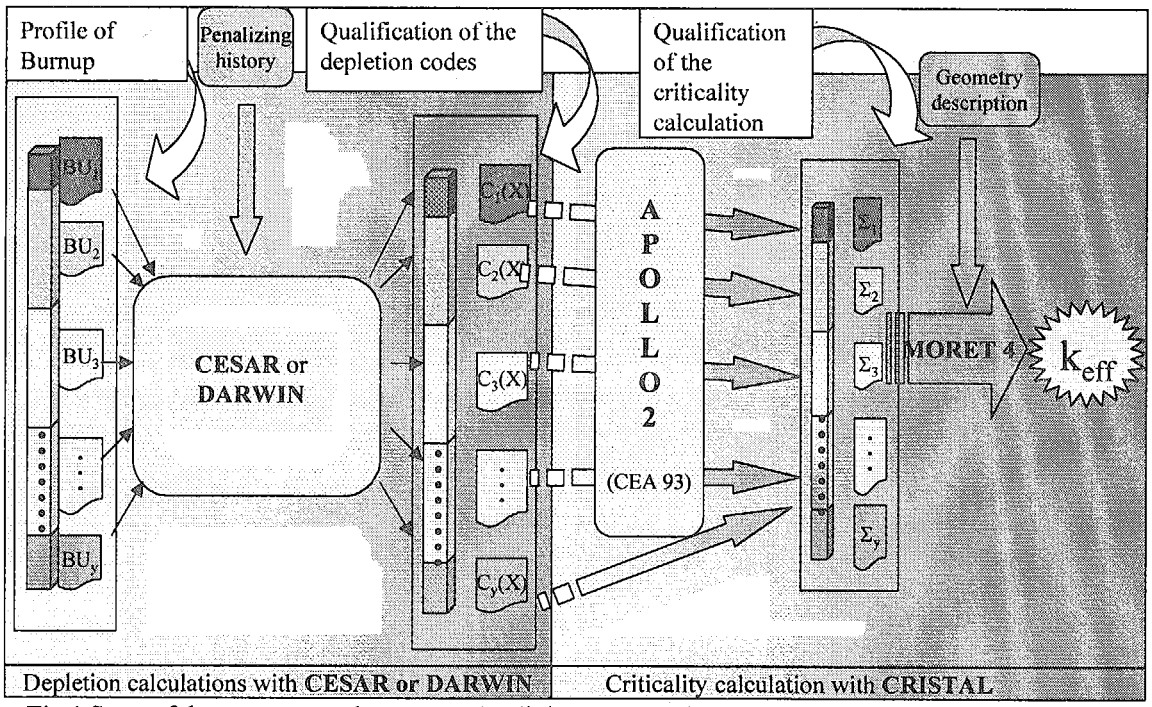

Fig.1 Steps of the process to take Burnup Credit into account in a criticality calculation with CRISTAL

The following isotopes have been studied: ^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{152}Sm and ^{155}Gd. It tests the capacity of the codes to calculate some critical situations representative of dissolution and storage. The completion of the Valduc program will make it possible to determine whether some correction factors have to be considered for the cross-sections (for that purpose the corrections factors will be applied to the concentrations in the criticality studies) of the fission products (possibly, in order to take into account an over-estimation of the absorption of a given fission product) or, eventually, if a margin must be added to reduce the value of the keff limit in the criticality studies.

Another program was led out at CEA/Cadarache, divided into two parts. The former one is devoted to fuel inventory validation by chemical analyses and microprobe measurements of PWR pins. The latter one, involving oscillation experiments, is related to the reactivity effect of the different nuclei responsible for the burnup credit. It validate the cross section of every individual fission product for 3 different types of spectrum. This experiments performed at MINERVE [20] showed that the calculated cross sections were in good agreement with the measured ones or that the cross sections of thermal absorbers tended to be underestimated (which is satisfactory in the point of vue of safety), except for the ^{133}Cs (the calculated value is up to 4 % greater than the measured one), for the ^{103}Rh (the calculated value is up to 13 % greater than the measured one), and for the ^{99}Tc (the calculated value is up to 4 % greater than the measured one).

2.3.3 Loosely coupled units

In the case of storage or transport of irradiated fuel, the level of the reactivity is mainly due to the edges of the assemblies, which are the least irradiated areas.

It raises the problem of the calculations of loosely coupled units with Monte Carlo codes; a special Working Group has been created at the OECD to study this problem.[21] Meanwhile different statistical methods have been implanted in the CRISTAL Package such as (i) the super history powering, (ii) the fission matrix (kij matrix) method, (iii) the stratified sampling ; those methods are being studied [22] to solve this particular type of problem.

Until the completion of the studies in that field, the R&D studies should be achieved with the following prescriptions:

- at the beginning of the calculations a certain amount of neutron sources has to be distributed in all of the different fissile units (the sources can be shared so that every fissile unit will include the same amount of starters at the first stage of the calculations),
- at the beginning of the calculations, some additional neutron sources will be placed at the top and the bottom of the fissile assemblies.

3. Gain Estimation

The comparisons between the keff value obtained with the actinide-only method and with a "new approach" [7] could be presented as: Δk = k(current method) - k(new approach with control rods inserted). The values of Δk for different configurations of storage (e. g. interim storage at La Hague) and transportation (e. g. TN17-2) are given below. For those calculations 17 axial zones were used. Some very pessimistic correction factors were used for the fission products and for the actinides to take into account, (i) the discrepancies observed between the results of the depletion calculations and the experimental data, (ii) the differences in the cross sections (condensate to 1 group of energy, for a given spectrum) noticed during the MINERVE program. Moreover, the pessimistic assumptions, regarding the conditions of irradiation described in paragraph 2.1.1., were taken into account.

Table 2 Comparisons between the keff value obtained with the actinide-only method and with the new method ($\sigma < 0.1$ %)

	BU and type of profile used in the new method	Δk = k(current method) - k(new method with control rods inserted)
with no cooling time	11 GWd/t penalizing profil	1.8 % to 2.6 %
	11 GWd/t standard profil	1.1 % to 1.5 %
	22 GWd/t standard profil	1.6 % to 2 %
	33 GWd/t standard profil	1.1 % to 1.8 %
	44 GWd/t standard profil	1.2 % to 1.5 %
with 5 years cooling time	22 GWd/t standard profil	2.9 % to 3.3 %
	33 GWd/t standard profil	3.3 % to 3.9 %
	44 GWd/t standard profil	3.2 % to 3.9 %

Finally, we can notice that, even if the correction factors are pessimistic and the irradiation history is very conservative and no realistic, the new method already gives a gain up to Δk = 3.9 %. This gain will be increased if the presence of control rods during the whole irradiation can be excluded : with no cooling time, a standard profil and for a burnup of 44 GWd/t, the gain will be up to Δk = 6 %.

4. Conclusion and Prospects

The objective of the Working Group is to study the conservatism of all steps of the process to take fuel burnup into account in the criticality-safety studies,

considering fission products and axial burnup profile.

The next studies are performing on :

- the possibility to exclude the addition of different conservatisms,
- the influence of the irradiation history on Eu^{154}, Cs^{134}, Cs^{137} and Nd^{148} in order to study the effect on the Burnup measure precision,
- the influence of the burnable poisons used in France,
- the effect of an horizontal profile of burnup and the control rods insertion histories,
- the determination of correction factors,
- the variation of the fuel density during the irradiation and its effect on the reactivity,
- the "stretch-out" effect.

The conclusions presented are related to the present state of knowledge and may differ from the final conclusions of the Working Group. The Working Group is still working on the effect of the parameters of the irradiation history and the gain estimation.

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