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SUMMARY REPORT ON THE FRENCH BURNUP CREDIT WORKING GROUP KNOWLEDGE REGARDING PWR UOX FUEL

French Burnup Credit WG

Rapport n° PSN-EXP/SNC/2017-177

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Pôle sûreté des installations et des systèmes nucléaires
Service de Neutronique et des risques de Criticité

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ABSTRACT

The "Burnup Credit" Working Group was established in 1997 to examine the various parameters, such as the irradiation conditions, the burnup profile and the nuclides (actinides and fission products), to be taken into consideration in the criticality studies that take credit from burnup.

This report offers an overview of the work that has been completed or agreed under this framework. It presents the group findings on the following topics:

- the axial distribution of nuclides or the axial burnup profile;
- methods for validating the actual burnup and its axial distribution;
- the calculation of nuclide concentrations after irradiation;
- the calculation methods that will be used to determine the effective multiplication factor for systems containing used fuel assemblies.

This document gathers together the work carried out by the French Burnup Credit Working Group; it is not a guide validating a particular method for taking burnup credit into account. All of the findings presented here may serve as a basis in industry for defining a method to take account of burnup credit in criticality studies; any industrial body effectively adopting such a method will also be responsible for defining it, based on its knowledge of the used fuel assemblies and the configuration to be addressed.

This document forms a collection of the work completed by the Working Group up to 1 January 2007 but does not necessarily reflect ongoing work in the various institutes.

HISTORIQUE DES MODIFICATIONS/*CHANGE HISTORY*

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CONTENTS

This report results from the work conducted by the members of the French Burnup Credit Working Group between 1997 and 2007. The English translation was checked by the members between 2015 and 2016.

The Working Group would like to take this opportunity to mention Mister Didier Biron, who worked hard to increase knowledge on burnup credit in France. The Working Group wishes to pay tribute to him through this report.

1 INTRODUCTION

In the early 1980s, a method was developed to take account of the burnup[1] to enable criticality studies to consider the decrease in reactivity, due to the burnup of uranium oxide-based fuels irradiated in light water reactors; this method was initially implemented to demonstrate the safety of the dissolution of highly enriched (4.4 wt%) used fuels at COGEMA UP2-400 plant at La Hague, and was subsequently extended to assembly transportation and to the design of the UP3 and UP2-800 plants.

This method, which was accepted by the French Safety Authority, takes into account the disappearance of uranium 235 and 238, and the appearance of the following actinides: uranium 236 and plutonium 238, 239, 240, 241 and 242. The burnup used in the studies must be lower than or equal to the mean burnup measured operationally over the 50 least irradiated centimeters of the assembly.

Given the increase in the initial enrichment of PWR Uranium Oxide (UOX)-based fuels, coupled with the considerable interim storage needs for used fuel assemblies, it was deemed necessary to define a method to take account of the burnup that would be less penalizing than that used previously; this would be approached by considering hypotheses closer to the physics of the phenomena (burnup profile modelling, consideration of fission products, etc.), but with conservatism to guarantee the conservative nature of the results.

In order to examine the various parameters of such a method, a Burnup Credit Working Group was established in 1997, bringing together EDF, FRAMATOME, CEA, COGEMA, SGN, TRANSNUCLEAIRE and IPSN. The findings presented in this report derive from the discussions and studies conducted by this group, along with the international publications produced in this field.

The aim of this document is to present all of the aspects to be addressed as part of this new approach, together with the resulting studies; it also provides an overview of the work already completed and the tasks underway.

The consideration of BurnUp Credit (BUC) is based on the modelling of the different physical phenomena associated with irradiation. These physical phenomena depend on the appearance and disappearance of nuclides, during the irradiation of the assemblies in the reactor when subjected to a variable neutron flux, and during cooling after fuel irradiation. Consequently, a BUC approach is structured around the following points:

[1] The burnup corresponds to the energy produced by a fuel per mass unit. This value is typically given in MWd/MTHM or GWd/MTHM (where MTHM denotes metric tons of initial heavy metal).

- the calculation of the concentrations of the different selected nuclides right at the end of irradiation or after a given cooling time;

- the axial and radial distribution of these nuclide concentrations, or the axial and radial burnup "profile";

- the guaranteed irradiation level of the assemblies and the axial burnup distribution;

- finally, the calculation of the effective multiplication factor for a given burnup profile and the associated nuclide concentrations.

Each of the above steps must be addressed in a way that guarantees conservatism when taking burnup credit into account, and will be covered by a dedicated chapter:

- Chapter 2 discusses the different axial burnup profiles that might be used in the studies; for a given mean burnup, it examines the impact of the axial burnup distribution along the assembly and describes the validation methods studied by the Working Group, demonstrating that the profile chosen for the criticality studies is penalizing compared to the actual axial burnup profile;

- Chapter 3 is devoted to the analysis of parameters for assessing the nuclide concentrations in the used fuel for a given burnup and cooling time; the conservatism of the concentrations obtained is associated with the irradiation and cooling histories selected for the calculations, as well as the results from the validation of the depletion codes used;

- Chapter 4 focuses on the sequencing of the criticality calculations taking burnup credit into account; in particular, it describes how a given discretized burnup profile can be used to:

    - introduce the nuclide concentrations for the various selected nuclides, for each established burnup zone (taking into account the calculation methods used to determine their concentrations);

    - calculate the effective multiplication factor; in this context, different calculation configurations were studied, including the configuration for an isolated assembly (this configuration can be used to assess the intrinsic effect of taking burnup credit into account for an assembly), interim storage configurations in normal and accident conditions (in pools at La Hague), and transport configurations;

Chapter 4 then presents the work carried out on the validation of cross-section libraries and criticality calculation codes for the consideration of burnup credit. The loose coupling and source convergence issues for the Monte Carlo calculations are also discussed in this chapter;

- finally, the conclusion reviews all of the results obtained, the studies currently in progress and future actions to be considered.

This document gathers together the work carried out by the French Burnup Credit Working Group up to 1 January 2007; it is not a guide validating a particular method for taking burnup credit into account.

The findings presented here may serve as a basis in industry for defining a method to take account of burnup credit in criticality studies; any industrial body defining such a method must take full responsibility for its development, giving consideration to the configuration to be addressed, the analyzed scenarios and its knowledge of the used fuel assemblies.

2 CONSERVATIVE BURNUP PROFILES

The studies conducted in the framework of the OECD, described in reference documents 1, 2, 3, 4, 5 and 6, revealed that the hypothesis of considering a uniform burnup across the entire assembly, and equal to the mean burnup, is not conservative when the average burnup exceeds 30 GWd/MTHM.

Indeed, for some configurations, typically with high irradiation rates, the ends of the assembly become considerably under-irradiated and govern the system overall reactivity level.

The work carried out by the Burnup Credit Working Group on determining an axial burnup profile was structured around two lines of research:

- the first involved initially determining axial burnup profile families, based on either measurements taken in the reactor or burnup profile calculations with three-dimensional codes, and then defining penalizing profiles for each defined profile family; this work, based on calculated profiles, is described in reference document 15;

- the aim of the second was to classify the actual axial burnup profiles measured in the La Hague plant into different families.

From these studies, along with the discussions held by the Burnup Credit Working Group, it was found that:

- the systematic use of a penalizing (perturbed) profile could lead to excessive conservatism (for example, the perturbed profile studied in reference document 14 leads to a conservatism of up to 11,000 pcm in Δk) and the conservative nature of the selected profile is, in any case, difficult to establish for all types of core management. However, validated approaches based on core calculations, using measurements taken in the reactor (not yet explored by the Working Group), could provide a possible alternative and be the subject of work presented in a later version of this report;

- the presentation of 3,000 17x17 PWR profiles measured at La Hague showed that there was little difference between the normalized profiles of the used fuel assemblies arriving at La Hague for any given burnup range (see report in reference 10).

Based on an experimental approach, one possible method, presented at the Burnup Credit Working Group meeting on 12 March 2001 (see reference 7), would involve considering an axial burnup profile similar to the standard one, and making measurements to check that the measured burnup at each point is actually higher than the burnup used in the studies.

As soon as the axial burnup profile is validated by a measurement, any profile can be selected for the criticality studies. However, if the chosen profile is required to cover the majority of the assemblies, it will need to be defined based on the profiles observed to date.

It will then be possible to justify the criticality safety of the storage or transportation of particularly perturbed profiles, which are not covered by a "standard" profile, by conducting a specific study:

- either by defining a very penalizing profile;
- or by reducing the burnup used in the study.

Finally, in some cases (such as the used fuel pits at La Hague), where the operator needs to guarantee only a minimal burnup (less than one irradiation cycle), it would be useful to define a method that accounts for the burnup without the needs for a systematic measurement of either the profile or the mean burnup.

The following points are presented in turn in the remainder of this chapter:

- possible approaches to the creation of an axial profile and the discretization selected for it;
- studies relating to a possible horizontal burnup gradient;
- the validation methods explored.

2.1 CHOICE OF THE BURNUP PROFILE FOR THE STUDIES

Two profile types are presented below. These relate to:

- the "standard" profiles, which are likely to be encountered for the majority of used fuel assemblies;
- the specific profiles, which are likely to be seen in very exceptional circumstances or to be used in a penalizing way.

2.1.1 GENERIC STUDIES (APPROACHES BASED ON FEEDBACK)

The members of the Working Group considered it useful to define a profile for each studied burnup range, based on the series of profiles observed at La Hague, which could then be used systematically for criticality studies implementing used fuel assemblies.

The observations made at La Hague showed that the normalized burnup profiles[2] were similar within the following three burnup ranges: [10 GWd/MTHM; 20 GWd/MTHM], [20 GWd/MTHM; 30 GWd/MTHM] and over 30 GWd/MTHM, with a flattening of the profile in the mid-section and a rise in the slope at the ends, when the burnup increases (see references 9 and 10).

Given these observations, it appears that a standard burnup profile can be defined for each burnup range, which covers most of the cases met and that its conservativeness can be demonstrated (for example, by making measurements to check that the actual burnup profile is always within the range of curves currently observed).

Several approaches may be considered, the principles of which are detailed in Appendix 8.

An approach suggested by IRSN (see reference 8) and illustrated in Figure 1 below, involves defining a conservative profile. A burnup profile may be deemed conservative if, on any part of the used fuel assembly, the burnup is lower than that of the actual profile. It should be noted that such a profile is not a normalized one (the mean is less than 1), which is penalizing.

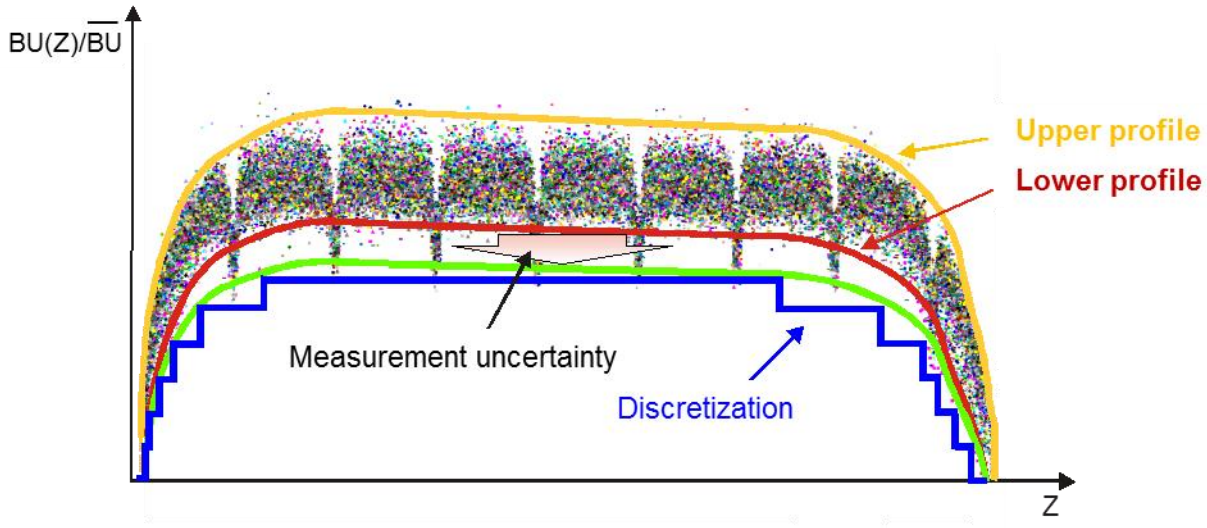


Figure 1: Principle for determining an overall profile from ^{137}Cs profile measurements for 1300 MWe PWR assemblies

Another approach was also examined by EDF (refer to Appendix 8 and reference document 9) to suggest a "Best Estimate" profile that would be representative of the majority of assemblies, based on the profiles observed at La Hague. This profile is not intended to be

[2] The normalized profile is defined as follows: ($P(z) = \frac{BF(z)}{\int_H BF(z)\,dz \;/\; H}$).

conservative but could be used at a later stage as part of a specific BUC approach. Details of this point will be provided in the next revision of this document.

COGEMA gave the Working Group a database containing the measurements (taken in the T1 burnup pit at the plant at La Hague) for the axial burnup profile of 644 used fuel assemblies in EDF 1300 MWe reactors, grouped into two series, named R103 and R202, and reprocessed at La Hague. COGEMA provided the gross count rates for ^{134}Cs and ^{137}Cs at centimetre intervals along two opposite sides (named route 1 and route 2 throughout the remainder of this document) of each of the assemblies and, deduced from these measurements, the mean burnup for each assembly and each side.

The majority of the assemblies reveals an initial ^{235}U enrichment of 3.10 wt%. The other assemblies, with an initial ^{235}U enrichment of 1.50 wt%, 1.80 wt%, 2.40 wt% and 2.95 wt%, are those from the first cores. These are associated with a fuel management with a balance enrichment of 3.10 wt%.

Over 94% of the assemblies had an average burnup of between 30 and 40 GWd/MTHM. The other assemblies had an average burnup of between 10 and 30 GWd/MTHM.

Table 1 below shows the distribution of the assemblies according to their mean burnup, for each of the two series and each of the two routes.

Table 1: Population of assemblies based on their burnups for the series and routes studied

	Number of assemblies			
	Series R103 Route 1	**Series R103 Route 2**	**Series R202 Route 1**	**Series R202 Route 2**
BU < 25 GWd/MTHM	10	14	13	13
25 < BU < 30 GWd/MTHM	6	3	7	34
30 < BU < 40 GWd/MTHM	290	289	318	291
BU > 40 GWd/MTHM	0	0	0	0
Total assemblies	**306**	**306**	**338**	**338**

From the data provided by EDF/DPI/DPN/UNIPE, it can be seen that 33% of the measured assemblies were irradiated in the final cycle under control rods (R or G) and that 10% were irradiated under control rods for over two thirds of the time spent in the reactor. The

database therefore contains a large number of assemblies placed under control rods during the irradiation process.

The representativeness of this database (assemblies with an initial ^{235}U enrichment equal to 3.1 wt% and a burnup of between 30 and 40 GWd/MTHM) remains to be confirmed for other burnups and enrichments, by providing axial burnup curve measurements for additional assemblies.

In order to compare the axial burnup curves, the initial data were formatted in advance. At each axial point (at centimetre intervals) of the active height of the assembly, the local count rate was divided by the total count rate and then multiplied by the height of the assembly to obtain a curve "centered" around the mean value 1. For each route, after removing the inconsistent points (negative or close to 0), and for each point, the measured local burnup was divided by the mean burnup value of the route studied, and by the number of measurement points.

The analysis of the formatted data finally revealed:

- a slight drop for the assembly profiles with a low burnup (less than 20 GWd/MTHM). It was also noted that the burnup values at the ends of the assemblies were lower for lightly irradiated assemblies than for moderately or highly irradiated assemblies. The figure below illustrates this phenomenon. The axial point is measured from the bottom of the assembly;

- consistent burnup measurements between routes 1 and 2 (profile and burnup): the relative mean difference between the burnups for routes 1 and 2 did not exceed 4% for the selected fuel elements (the burnup uncertainty measurement had been calculated at 5%). This phenomenon implies an equivalent burnup on both sides of the assemblies;

- consistency between the assemblies burnup profiles, ranging between 30 and 40 GWd/MTHM. This observation enables us to conclude that the position of an assembly during irradiation, whether it is under a control rod or not, has no specific impact on its burnup profile when the assembly comes to the end of its life.

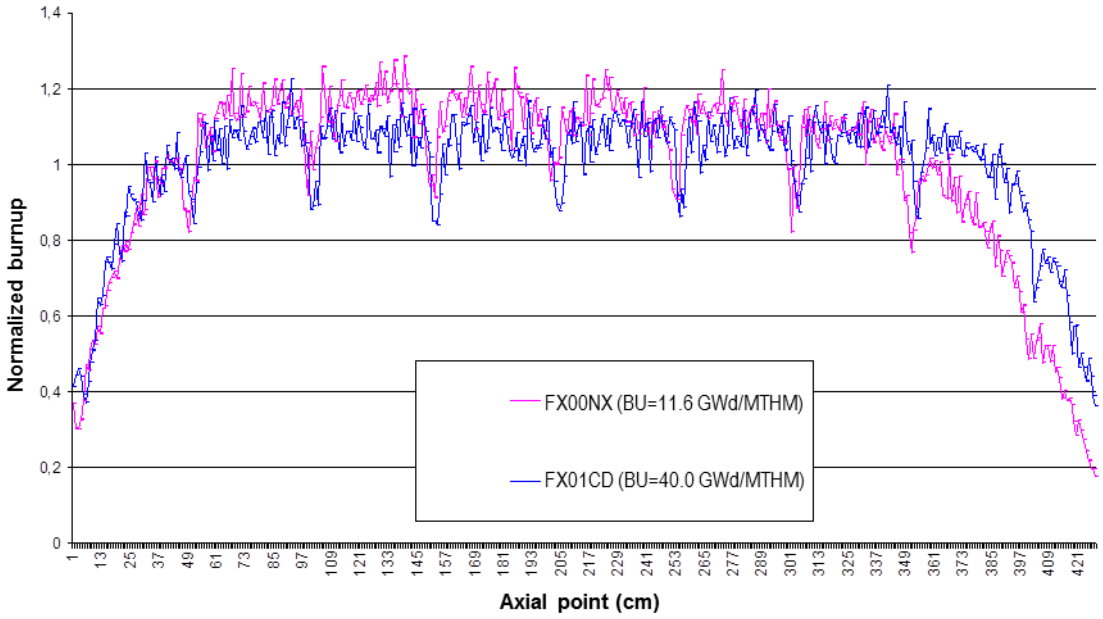


Figure 2: Normalized axial burnup profiles for two assemblies from the COGEMA database: FX00NX (BU=11.6 GWd/MTHM) and FX01CD (40.0 GWd/MTHM)

Based on this feedback, EDF suggested establishing an axial burnup profile representative of the assemblies contained in the COGEMA database with initial ^{235}U enrichment equal to 3.1 wt% and with a mean burnup of between 30 and 40 GWd/MTHM. To do this, at each axial point (totalling 427 for a 1300 MWe PWR assembly) an arithmetic mean is calculated based on all of the normalized profiles, corresponding to assemblies with an initial ^{235}U enrichment equal to 3.1 wt% and a mean burnup of between 30 and 40 GWd/MTHM. The mean profile obtained for each assembly is then normalized again. This profile may be considered as being a mean model of the profiles supplied by COGEMA.

To illustrate the low variability of the studied profiles, Figure 4 shows the normalized mean curves together with the maximum and minimum overall curves obtained for the set of assemblies with a mean burnup of between 30 and 40 GWd/MTHM. The differences between the overall curves and the mean curve are quantified in Appendix 8.

Table 2 and Figure 3 below show the mean profile obtained, discretized according to 12 zones.

Table 2: Description of the mean profile discretized according to 12 zones

Zone	Length (cm)	Beginning of zone	End of zone	Normalized BU
1	6	0	6	0.457
2	2	6	8	0.543
3	3	8	11	0.653
4	5	11	16	0.747
5	20	16	36	0.919
6	17	36	53	0.977
7	293.8	53	346.8	1.0618
8	35	346.8	381.8	1.014
9	17	381.8	398.8	0.935
10	20	398.8	418.8	0.736
11	5	418.8	423.8	0.545
12	3	423.8	426.8	0.476

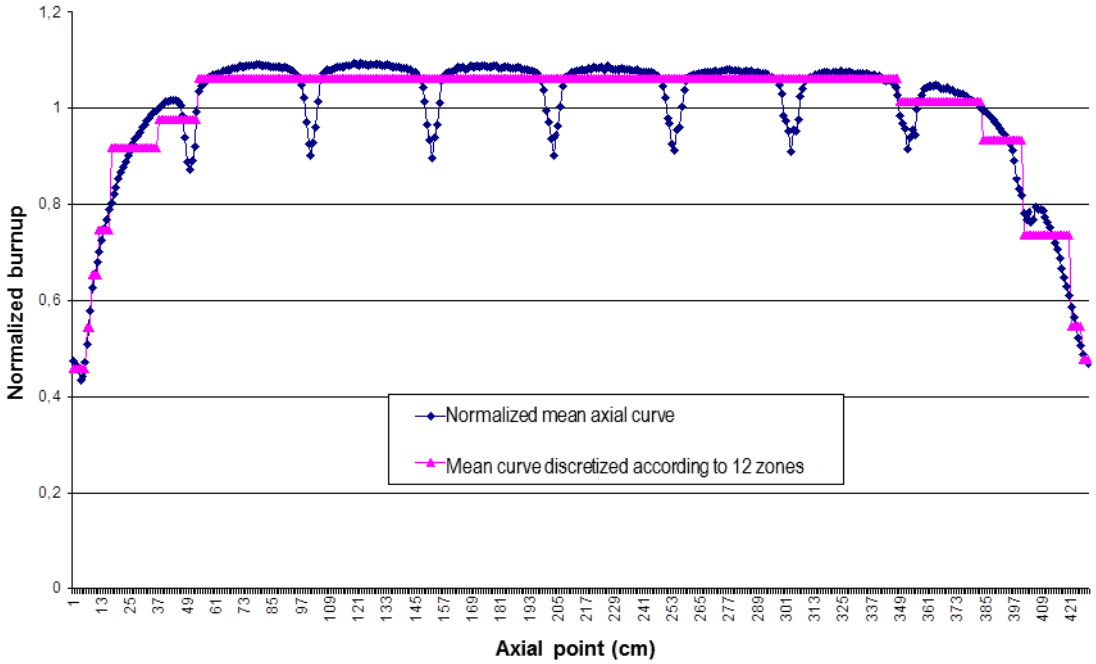


Figure 3: Normalized mean axial curves representative of an assembly with initial enrichment equal to 3.1 wt% and a burnup of between 30 and 40 GWd/MTHM from the COGEMA database, together with the mean curve discretized according to 12 zones

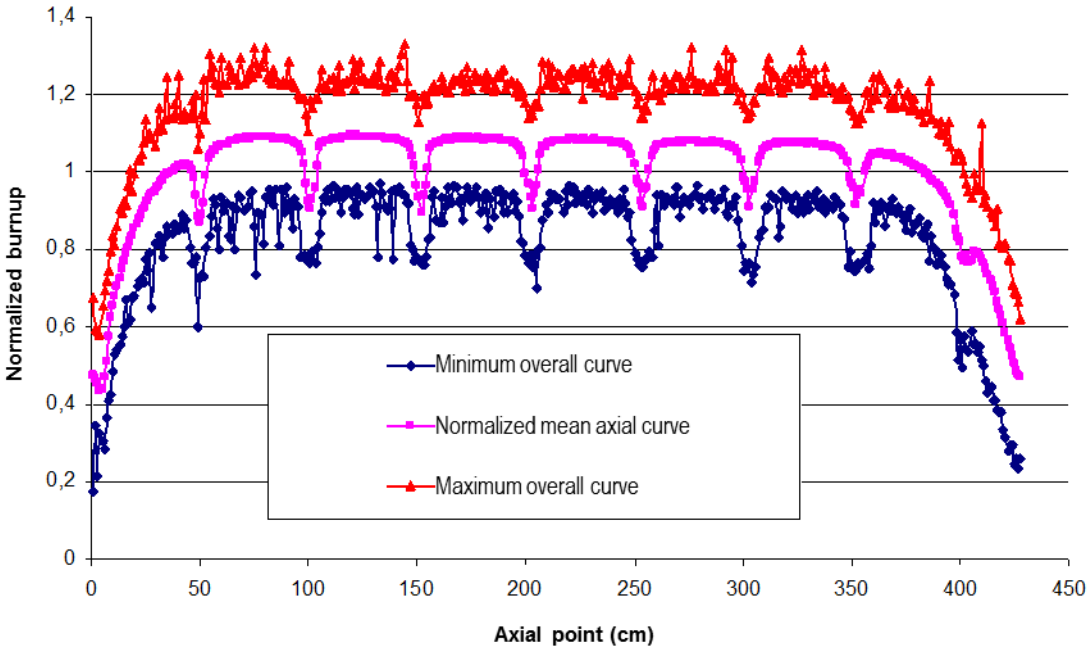


Figure 4: Normalized mean axial curves, and minimum and maximum overall curves, obtained from the assemblies with initial enrichment equal to 3.1 wt% and a burnup of between 30 and 40 GWd/MTHM from the COGEMA database

In conclusion, the statistical analysis of the burnup profiles supplied by COGEMA does not reveal any particular axial deformation attributable to operation with partially inserted rods. The mean profile obtained is representative of the curves likely to be observed in French PWR operating modes used currently, or envisaged in the near future, including for the EPR. This representativeness will be extended to other axial burnup curve measurements for fuel assemblies. If there is no significant change in the operating mode for these reactors, and since the profile obtained is not found to be highly variable, there does not appear to be any reason why this profile should not be used in the criticality safety studies that take burnup credit into account. However, the studies that use this profile will need to incorporate margins to ensure the overall conservatism of the approach.

2.1.2 SPECIFIC PROFILES

A specific method needs to be defined for the assemblies whose burnup profiles are not included within the range of curves observed at La Hague.

One of the options for taking "specific" cases into account would be to calculate a "penalizing" profile.

As an example, the calculations performed at EDF and presented in the report in reference 15, made it possible to create a particularly penalizing burnup profile (profile "no. 8"); using this highly perturbed profile can lead to a significant increase in reactivity

which, according to the study conducted by IRSN in reference 17 may reach 11,000 pcm (in Δk) for an assembly initially enriched at 4.5 wt% and irradiated at 44 GWd/MTHM.

In any event, the penalizing nature of the profile obtained using these specific methods must undergo validation. This point is addressed in chapter 2.3.

2.1.3 DISCRETIZATION OF THE SELECTED PROFILE

During the meeting of the Working Group on 12 March 2001, a conservative and penalizing profile discretization method was discussed. The smaller the number of zones leads to a more penalizing profile (meaning a higher k_{eff}). This methodology consists of taking the burnup in a particular fuel zone to be equal to the minimum burnup of the profile assumed for that zone. Such a breakdown therefore resulted in the creation of a profile that was not normalized. This hypothesis is shown in the figure below.

In this respect, the use of this undeniably conservative hypothesis does not lead to any notable overestimation of reactivity.

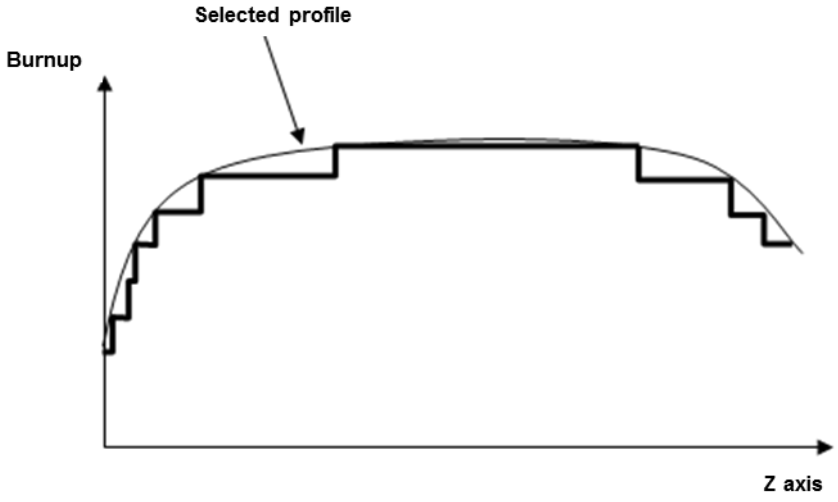


Figure 5: Discretization of the selected profile in 10 axial zones

Studies were also conducted to determine the appropriate number of axial zones (which would not be too penalizing in terms of the value of the effective multiplication factor and which would not require an excessive number of measurements).

Twelve zones appeared to be a reasonable option (the increase in the number of zones from 11 to 17 would lead to a gain of around 500 pcm); this choice is the result of an industrial decision, and means that if the calculation time is increased (due to an increase in the number of zones) then the value of the effective multiplication factor decreases. Several

types of axial breakdown were studied in the above-mentioned document, together with the breakdown corresponding to a constant burnup difference for each axial zone.

It should be noted that choosing a sensible number of zones can depend on the type of profile used.

2.2 RADIAL BURNUP GRADIENT

An assembly burnup has a more or less pronounced horizontal gradient, depending mainly on its position during the last cycle (at the periphery of the core, for example).

With regard to the spatial distribution of the nuclides in the used fuel assembly, studies have revealed, notably in the DOE document (reference 16), that differences can exist between the burnups on two opposite sides of a used fuel assembly; the calculated values in the aforementioned document are, as a maximum, approximately:

- 25% for burnups corresponding to 1 irradiation cycle;
- 15% for burnups corresponding to 2 irradiation cycles;
- 15% for burnups corresponding to 3 irradiation cycles.

Burnup measurements taken at COGEMA at La Hague for three orders of reprocessing (863 assemblies), on two sides of the fuel, revealed a maximum difference[3] of 22% between the two sides for a 1300 MWe PWR assembly with a burnup of around 34 GWd/MTHM (see minutes of meeting in reference 10).

The study conducted at IRSN and cited in reference 11 show that this radial burnup gradient represents an increase in reactivity of around 600 pcm for two assemblies placed side by side (with the most reactive surfaces against one another), with initial enrichment equal to 4.0 wt% and irradiated up to 33 GWd/MTHM. For interim storage configurations and transportation in a borated basket, this effect does not exceed 500 pcm.

However, in the case of four assemblies isolated in water, the use of a horizontal burnup gradient of 22%, modelled by two zones, leads to an increase in reactivity of around 650 pcm for one irradiation cycle and around 1,600 pcm for three cycles.

One should therefore remain cautious of the risk of a horizontal burnup gradient appearing. Indeed, such a burnup gradient may have a significant effect on the k_{eff} value of the studied

[3] Difference derived directly from the measured values without taking uncertainties into account.

configuration, particularly when the assemblies are neutronically coupled (in the absence of borated baskets, for example).

Three approaches may be considered to address this phenomenon:

- the first relates to the possible ways of measuring this horizontal burnup gradient;
- the second involves the possibility of systematically lowering the assemblies burnup to take this phenomenon into account in a very conservative way; in this case, the effect in terms of reactivity of using an actual gradient may be analyzed in order to quantify the conservatism associated with reducing the burnup;
- a final option would be to model the assemblies burnup gradient by considering, for example, the least irradiated surfaces against one another for the criticality calculations.

2.3 VALIDATION METHODS FOR THE BURNUP PROFILE SELECTED FOR THE STUDIES

The decision to take account of a burnup profile appears to be justified, subject to the validation of the actual burnup for the used fuel assemblies that will be stored or transported. The Working Group therefore deemed it necessary to study the available validation methods.

Initially, only the existing assembly burnup measurement methods were studied, together with their limitations. Other validation methods are currently being examined by particular members of the Working Group (EDF and AREVA-NP).

Burnup measurement methods are used to determine an assembly means burnup as well as the burnup profile. The approaches are as follows:

- to determine an assembly mean burnup, a measurement of the used fuel assembly neutron emission, a gamma spectrometry measurement or a total gamma measurement is taken;
- to measure the burnup profile, a gamma spectrometry measurement is taken.

The burnup measurement is based on the relationship between the emission (neutron or gamma) of certain nuclides and the burnup. It relies on an iterative process using the measured neutron (or gamma) radiation values of certain nuclides and the predicted (tabulated) values of these neutron (or gamma) emissions, based on the burnup, using a depletion code in conjunction with the measurement procedure.

The neutron measurement is used to determine neutron emission, basically due to the elements ^{242}Cm and ^{244}Cm, which is, as an initial approximation, proportional to the fourth power of the burnup. As the neutron beam is not collimated, this method can be used to determine an assembly mean burnup.

Gamma spectrometry is used to determine the relationship between the gamma emission of the elements ^{134}Cs and ^{137}Cs, which is directly proportional to the burnup. These are local measurements, which can also be used to determine the burnup profile, either by making repeated measurements of the ^{134}Cs/^{137}Cs ratio in several different places, or by making successive measurements of the axial profile of ^{137}Cs.

In the remainder of this chapter, by taking an approach based on the burnup profiles feedback, we suggest first studying the measurement uncertainties and then defining the measurement points that would need to be established based on the discretized profile used in the criticality safety study (derived from the approach presented in the previous chapter).

2.3.1 MEASUREMENT UNCERTAINTIES

Given the nature of the process used to determine the burnup, there are two sources of uncertainty associated with the measurements involved:

- uncertainties relating to measurement precision (in turn linked to the measurement method and the precision of the device used);
- uncertainties relating to the depletion calculation codes used to associate the burnup with the measured concentrations (or concentration ratio).

With regard to this last point, the uncertainties are linked to the modelling hypotheses used in the calculation codes (modelled relationship chains, resolution of equations, etc.), the source data (efficiency, cross-sections, etc.), and any influence that the irradiation conditions might have on the concentrations of the measured nuclides.

Moreover, with regard to an assembly mean burnup, it should be noted that the reactor operator provides a mean value, which could be used to reduce the measurement uncertainties.

In this context, it appears that the uncertainties associated with burnup measurement need to be analyzed. Details of this analysis are provided in Appendix 14.

- Absolute gamma spectrometry measurement of ^{137}Cs

  This type of measurement benefits from the fact that ^{137}Cs is radioactive for a very long time and that this nuclide is, therefore, barely affected if a cooling time error is introduced into the calculations. Furthermore, the fission yield from uranium is identical to that from plutonium. The mass balance of ^{137}Cs is thus proportional to the burnup and is independent of the assembly irradiation conditions.

  The measurement uncertainty depends on the quality of the calibration, the measuring conditions and the processing speed. It ranges from 15% to 25% and mainly results from the uncertainty of the relative position of the detector in relation to the assembly. This method has not been officially adopted at La Hague.

- Gamma spectrometry measurement of the $^{134}Cs/^{137}Cs$ ratio

  This has been the official method used for over 15 years to measure the assemblies in France and in other countries. It was first implemented in 1980 for CNA fuel, which had a cladding made of stainless steel.

  The benefit of this method (as compared to absolute ^{137}Cs measurement) is that it allows a self-calibration of the apparatus, and it is much less sensitive to the measurement conditions. However, the production of ^{134}Cs is highly dependent on the irradiation history.

  This method can be used to validate the burnup with an accuracy of approximately 15%.

- Gamma spectrometry measurement of ^{60}Co, ^{144}Pr or ^{154}Eu

  The measurement of ^{60}Co can provide feedback on the assemblies thermal output. This measurement is also used to position the assembly grids and the fissile column.

  The $^{144}Pr/^{137}Cs$ ratio can be used to estimate the cooling time for a used fuel assembly.

  The gamma spectrometry measurement of the $^{154}Eu/^{137}Cs$ ratio can also help determine the burnup, although with a higher level of uncertainty than with the $^{134}Cs/^{137}Cs$ method (due to uncertainties on the europium parent nuclides cross-sections).

- Passive neutron measurement

  A gamma spectrometry measurement (for ^{134}Cs and ^{137}Cs) has negligible sensitivity from the fourth row of rods.

  Passive neutron measurement compensates for this shortcoming. It is very easy to perform but requires a validated depletion calculation code to determine the in-line production of curium (or prior calculations for each assembly).

  This method yields low uncertainty values (of around 10%) for standard burnups for assemblies undergoing reprocessing ($\geq$ 3 cycles).

2.3.2 MEASURED VALUES TO BE USED

If we take a profile with N axial zones, we then need to check N+1 burnup values to guarantee that, at any point, the actual burnup is greater than the burnup considered in the study.

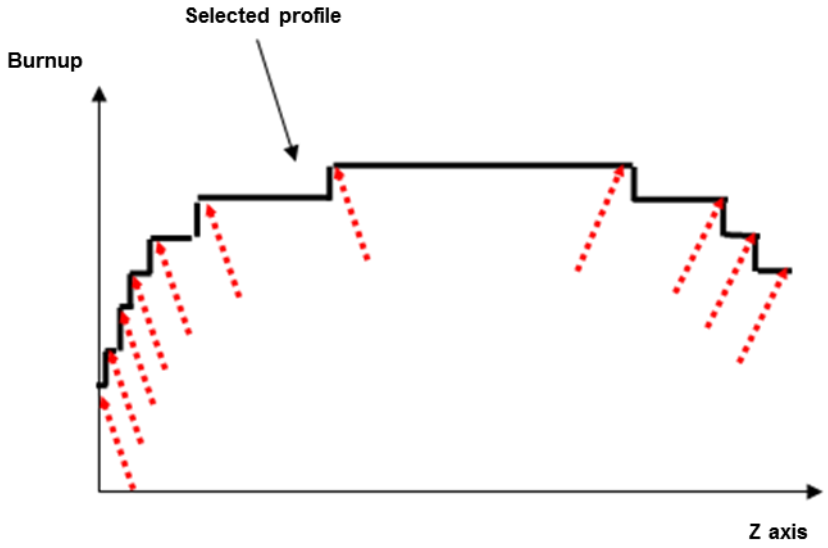


Figure 6: Burnup values that need to be guaranteed by the measurement

Checking the correspondence between the measured burnup and that selected for the study, at each and every point, is only a valid method if the measuring system gives the exact value of the burnup.

The experimental determination of the burnup, involving experimental measurement methods and providing calculation results, is complex and leads to uncertainties.

Therefore, if the operator needs to process assemblies with a mean actual burnup equal to BU_{actual}, then the criticality studies will be performed with a burnup BU_{study} equal, at most, to BU_{actual} x (1 - i), where i is a coefficient linked to the uncertainty of the measurement (it should be stated whether this uncertainty corresponds to 1, 2 or 3 σ); the actual mean burnup of the assembly will be greater than the mean burnup in the study, taking into account the uncertainties associated with the measurement and the method used to discretize the axial profile. This point is shown in the figure below.

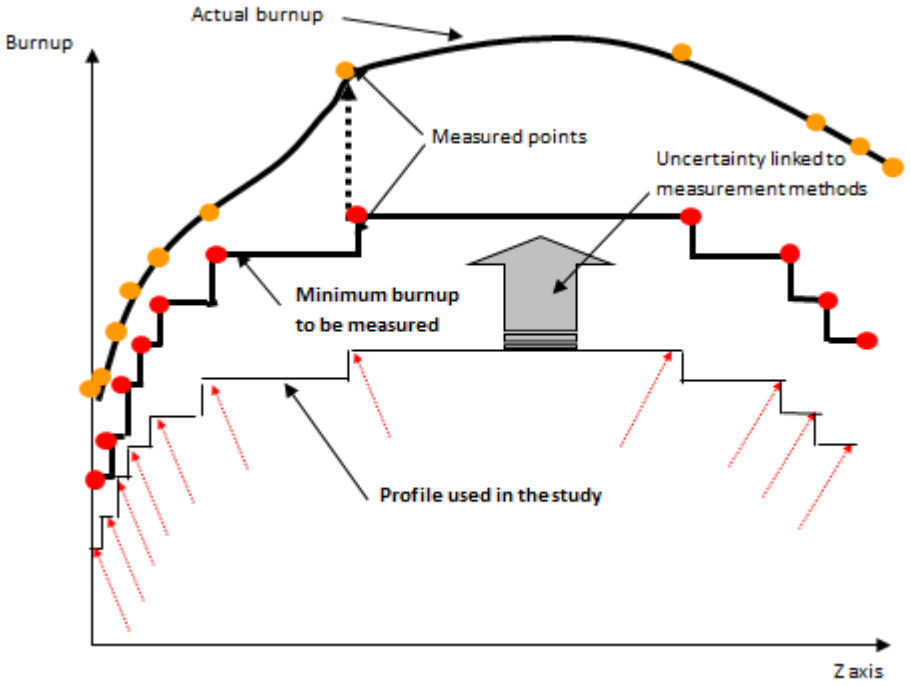


Figure 7: Incorporation of measurement uncertainties

Finally, it was noted that if the burnup used in a study is much lower than the assemblies actual burnup, then the number of measuring points can be reduced. This is illustrated in the figure below.

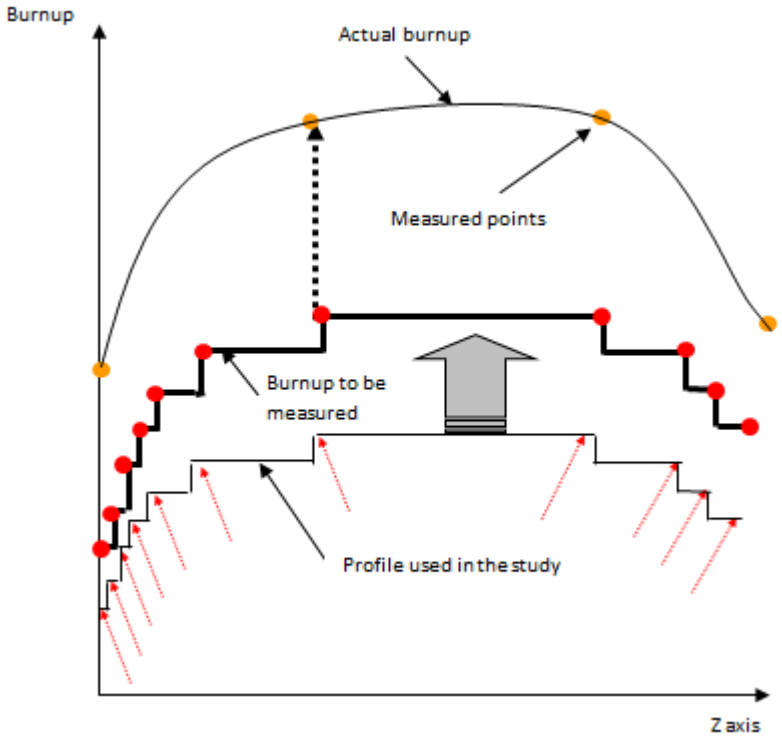


Figure 8: Case in which the burnup needed to justify a configuration is much lower than the actual burnup - 4 measuring points needed

2.4 CONCLUSION ON THE POTENTIALLY USABLE PROFILES AND VALIDATION METHODS

The Working Group is currently conducting studies to determine a burnup profile, based on the range of used fuel profiles measured at La Hague, that could be used generically with the vast majority of the used fuel assemblies. This profile will be provided for a burnup range and must undergo validation (by measurement, for example).

For assemblies with a burnup profile that is likely to be very specific (assemblies subject to refined profiles), or when no systematic measurement is adopted, a specific process can be defined. This process must guarantee the conservatism of the approach (by considering sufficiently low burnup values, for example).

Whenever a "standard" burnup profile is used in a study, several types of model may be considered. For example, penalizing discretization involves taking the burnup in a particular zone to be equal to the minimum burnup in that zone. Each industrial body can determine the optimal number of zones (in terms of the increase in reactivity or calculation time) to select for the studies.

It also seems necessary to ensure that the mean burnup value and the axial burnup profile are reliably determined. To date, the only method explored by the Working Group to guarantee the axial burnup distribution involves measurement. If the correspondence between the profile used in the studies and the actual burnup profile is checked by a measurement, this must guarantee in particular that, at any point, the actual burnup is greater than the burnup used in the study. In certain cases, where the operator does not wish to take any measurements, the use of a penalizing profile may be considered. However, the conservatism of the choice of profile must be provided by other profile validation methods.

3 NUCLIDE CONCENTRATIONS IN THE USED FUEL ASSEMBLIES

Burnup credit is the negative reactivity caused by the burnup of the fuel when it is irradiated in the reactor core. This increase in negative reactivity corresponds to the appearance and disappearance of different nuclides during the irradiation process. It therefore appears necessary to consider all of the fissile elements produced during irradiation (those playing an appreciable role in the chain reactions) and desirable to take account of the largest number of absorbent nuclides produced during irradiation (actinides and fission products) whose presence in the fuel is certain (without considering volatile nuclides or those with too short decay time compared to the cooling times proposed for the study).

The actinides and fission products selected for the OECD benchmarks are as follows:

- Actinides: Uranium 234, 235, 236 and 238, Plutonium 238, 239, 240, 241 and 242, Americium 241 and 243, and Neptunium 237;
- Fission products: Samarium 149 and 152, Rhodium 103, Neodymium 143, Cesium 133 and Gadolinium 155, in addition to ^{101}Ru, ^{95}Mo, ^{99}Tc, ^{147}Sm, ^{150}Sm, ^{151}Sm, ^{145}Nd, ^{109}Ag and ^{153}Eu; these 15 fission products are responsible for approximately 80% of the negative reactivity caused by all of the fission products (according to reference document 12).

The first six fission products listed above (^{149}Sm, ^{152}Sm, ^{103}Rh, ^{143}Nd, ^{133}Cs and ^{155}Gd) are alone responsible for 50% of the negative reactivity introduced by all of the fission products. The choice of these six fission products was justified for the first time at the ICNC at Oxford in 1991 (see publication reference 13). This choice was based on the following criteria:

- minimising the number of fission products (FPs) and, therefore, the associated problems, while maintaining a large proportion of the FPs total absorption;
- disregarding the gaseous and volatile bodies;
- retaining the stable bodies, or those with a daughter product that has greater absorption than the parent;
- minimising the non-solubility and precipitation problems in the acid phase, more specifically for dissolution.

The choice of the 15 fission products was based on the same types of criteria as those listed above.

The following table presents an estimate of the negative reactivity provided by each nuclide for a PWR UO_2 fuel, with initial ^{235}U enrichment equal to 3.3 wt%, subjected to irradiation up to 40 GWd/MTHM followed by five years of cooling.

Table 3: Negative reactivity in pcm ($10^5.\Delta k/k$) of burnup credit nuclides (initial enrichment = 3.3 wt%, burnup = 40 GWd/MTHM, cooling time = 5 years)

Actinides	Negative reactivity (pcm)	Fission products	Negative reactivity (pcm)
^{234}U	120	^{143}Nd	900
^{235}U		^{145}Nd	410
^{236}U	910	^{147}Sm	230
^{238}U		^{149}Sm	1030
^{238}Pu	310	^{150}Sm	270
^{239}Pu		^{151}Sm	500
^{240}Pu	8370	^{152}Sm	490
^{241}Pu		^{95}Mo	290
^{242}Pu	710	^{103}Rh	1360
^{237}Np	620	^{155}Gd	1500
^{241}Am	1290	^{153}Eu	390
^{243}Am	280	^{109}Ag	250
		^{99}Tc	440
		^{101}Ru	220
		^{133}Cs	750

However, it should be pointed out that the use of these fission products and actinides in the criticality calculations means that it is necessary to be able to accurately estimate their abundance after irradiation and to validate the cross-sections of these nuclides in configurations close to the actual ones. **This constraint may lead the criticality expert to exclude some of the nuclides listed above.** Thus, given the lack of critical experiments with fission products, the Americans currently limit themselves to actinides when taking burnup credit into consideration.

As soon as the choice of nuclides is finalized, it must be ensured that the depletion calculations, which provide the composition of a used fuel, produce conservative results in terms of used fuel composition. This conservatism depends on two factors:

- the accurate description of the irradiation conditions of each modelled assembly or the definition of a "penalizing" irradiation history for all of the assemblies. It seems obvious that accurately describing the irradiation and cooling conditions is not compatible with conducting generic studies for a given type of assembly; the Working Group has, therefore, been interested in studying irradiation and cooling conditions;

- the validation of the chosen depletion codes in the selected field of use (possible use of correction factors to ensure the conservatism of the calculated used fuel composition).

The following paragraphs are devoted to describing the selected approach for each of the two points above.

3.1 IRRADIATION HISTORY

For a given burnup, the irradiation conditions may have a significant effect on the concentration of the selected nuclides. Indeed, according to the reports produced by the Burnup Credit Working Group in references 14 and 15, together with the DOE report in reference 16, it appears that the fuel reactivity increases when the irradiation conditions lead to:

- a neutron spectrum hardening: this may be linked to the presence of control rods in the assembly, the increase of the boron concentration, the increase of the moderator temperature, the presence of MOX fuel assemblies close to the assembly or the presence of burnable poisons;
- a decrease in the irradiation time which, for a given burnup, is associated with an increase in the specific power;
- an increase in the temperature of the fuel.

The remainder of this paragraph will therefore present:

- first, a description of the physical mechanisms that cause the variations in the concentrations of certain nuclides when an irradiation parameter varies;
- second, the Working Group recommendations with regard to the irradiation conditions likely to modify the nuclide concentrations of a used fuel assembly.

It will also address the phenomena associated with the cooling time, along with the cooling time ranges likely to be used in the studies, even though they do not, strictly speaking, form part of the irradiation conditions.

3.1.1 ASSOCIATED PHYSICAL PHENOMENA

In the remainder of this paragraph, the physical phenomena that lead to the modification of the nuclide concentrations obtained after irradiation are described, in particular:

- the influence of the burnup on the nuclide concentrations for the selected nuclides;
- the effect of neutron spectrum hardening;

- the influence of the specific power on the used fuel composition;
- the role of the irradiation history, i.e. the length of the irradiation cycles or the presence of inter-cycles;
- the effect of the fuel temperature;
- finally, the mechanisms associated with the cooling time in the assemblies at the end of the irradiation cycle.

3.1.1.1 Influence of the burnup

Generally speaking, the nuclide concentration increases with the burnup for all considered nuclides, with the exception of the following three:

- ^{235}U and ^{238}U, which are initially present, and their concentrations decrease during irradiation;
- ^{149}Sm, which reaches a maximum concentration, then starts to decrease (the IRSN study in reference 17 shows that the maximum concentration is all the more delayed since the specific power is high).

In view of this last point, the conditions for including ^{149}Sm would need to be specified.

3.1.1.2 Effect of neutron spectrum hardening

If the irradiation conditions lead to neutron spectrum hardening, there will be fewer fission processes on the ^{235}U, while the captures on the ^{238}U will increase. These captures will prompt an increase in the production of plutonium (followed by a greater proportion of fission processes on the plutonium isotopes, which will harden the spectrum further).

With regard to the fission products, neutron spectrum hardening will lead to:

- a change in the distribution of the fission products, since a greater number of fission processes will occur on the plutonium isotopes, which do not have the same fission yields as ^{235}U;
- a smaller number of absorption processes associated with the "thermal" fission products (such as ^{149}Sm, ^{143}Nd and ^{155}Gd) during irradiation.

Finally, the calculations show that the reactivity associated with the used fuel is greater when the neutron spectrum is harder.

3.1.1.3 Effect of an increase in the specific power

For a given burnup, an increase in the specific power leads to a reduction in the irradiation time, which reduces the number of radioactive decay events. It is difficult, at first sight, to understand the effect of an increase in the specific power on the final concentration, since most of the elements are produced by radioactive decay and neutron absorption, but also disappear in accordance with these two phenomena.

A study was carried out at IRSN, as part of the work cited in reference document 17, on a fuel initially enriched at 4.5 wt%; this work presented the variation in the concentration of the different nuclides according to the specific power and the burnup. It emerged that an increase in the specific power leads to:

- a low variation in the concentration of actinides, except ^{239}Pu, ^{241}Pu and ^{242}Pu, whose concentrations increase when the specific power increases, and ^{241}Am and ^{238}Pu, which decrease;
- a very low variation in the concentrations of ^{99}Tc, ^{101}Ru, ^{109}Ag, ^{133}Cs, ^{143}Nd, ^{145}Nd and ^{153}Eu, and a very high variation in the concentrations of ^{95}Mo, ^{103}Rh, ^{151}Sm, ^{147}Sm, ^{149}Sm, ^{152}Sm, ^{155}Gd and ^{155}Eu.

3.1.1.4 Irradiation history

The irradiation history is linked to the variation in the specific power during the cycles and the incorporation of "inter-cycle" periods; according to reference document 18, these periods may lead to significant changes in the concentrations of certain nuclides.

This is because the presence of a sufficiently long cooling period between cycles favours the creation of elements from radioactive decay events.

If these elements have a large capture cross-section for spectra that are representative of the neutron flux in the reactor, they will disappear when the irradiation process resumes.

As a result:

- For certain "thermal" absorbent nuclides (in particular ^{155}Gd and ^{241}Am), taking a cooling period into account after irradiation leads to a creation, due to radioactive decay and, consequently, the disappearance of their parent nuclides.

- Alternating between shutdown periods and operating phases may, in certain conditions, lead to the disappearance of thermal absorbents, without the concentration of the parent nuclides having the time to re-reproduce[4].

3.1.1.5 Effect of an increase in the fuel temperature

An increase in the fuel temperature causes the resonance of the capture cross-sections of ^{238}U to increase, which leads to greater absorption on this nuclide (Doppler effect). These absorptions on ^{238}U prompt an increase in the production of plutonium and, for a given burnup, a lower consumption of ^{235}U.

3.1.1.6 Cooling time

The IRSN calculations presented in reference document 17 showed that reactivity decreases until the cooling time reaches 100 years. The figure below is taken from reference document 17.

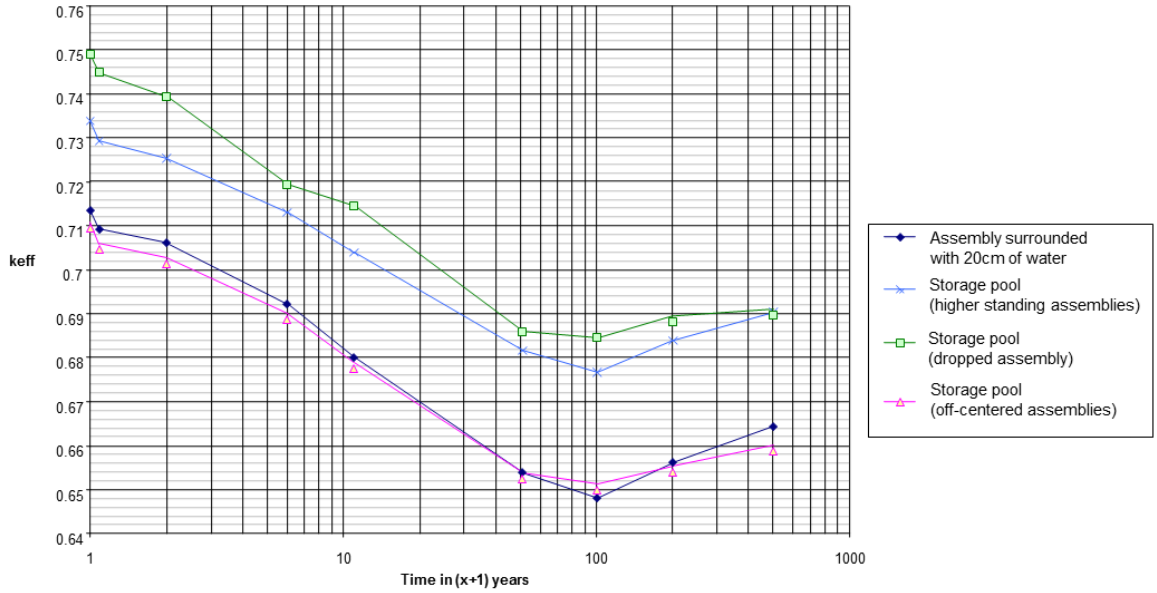


Figure 9: Effect of the cooling time on the effective multiplication factor after an irradiation of 44 GWd/MTHM (on an assembly initially enriched at 4.5 wt%)

This decrease in reactivity, during the first 100 years following irradiation, is linked to the increase in ^{155}Gd (whose concentration increases until the cooling time reaches 50 years) and the decrease in ^{241}Pu (together with the increase in ^{241}Am). The concentrations of ^{151}Sm and ^{238}Pu decrease, which should lead to an increase in reactivity. However, this has no

[4] The absorbent nuclides ^{149}Sm and ^{143}Nd, in the "thermal" domain, see their concentrations increase by 55% and 3% respectively during the first year of cooling, following an irradiation period of 44 GWd/MTHM (according to reference document 17). The concentration of ^{239}Pu also increases by 2%. Their concentrations will reduce with the resumption of irradiation.

predominant effect on reactivity (for cooling times of less than 100 years, only the concentrations of ^{241}Pu, ^{241}Am, ^{238}Pu and ^{237}Np vary to any significant degree).

For cooling times of more than 100 years (but less than 500 years), the reactivity increases slightly. This increase is mainly linked to the disappearance of ^{241}Am (which reduces by 14% during a cooling time of between 100 and 200 years). The number of ^{241}Am atoms decreases, just as the number of ^{237}Np atoms increases (but this nuclide has a smaller cross-section than ^{241}Am).

The criticality studies conducted by the ANDRA, to define a concept for irreversibly storing used fuel assemblies in deep geological layers without prior reprocessing, led EDF/SEPTEN to conduct a close analysis of changes in reactivity over periods of up to 5 million years (see reference document 19).

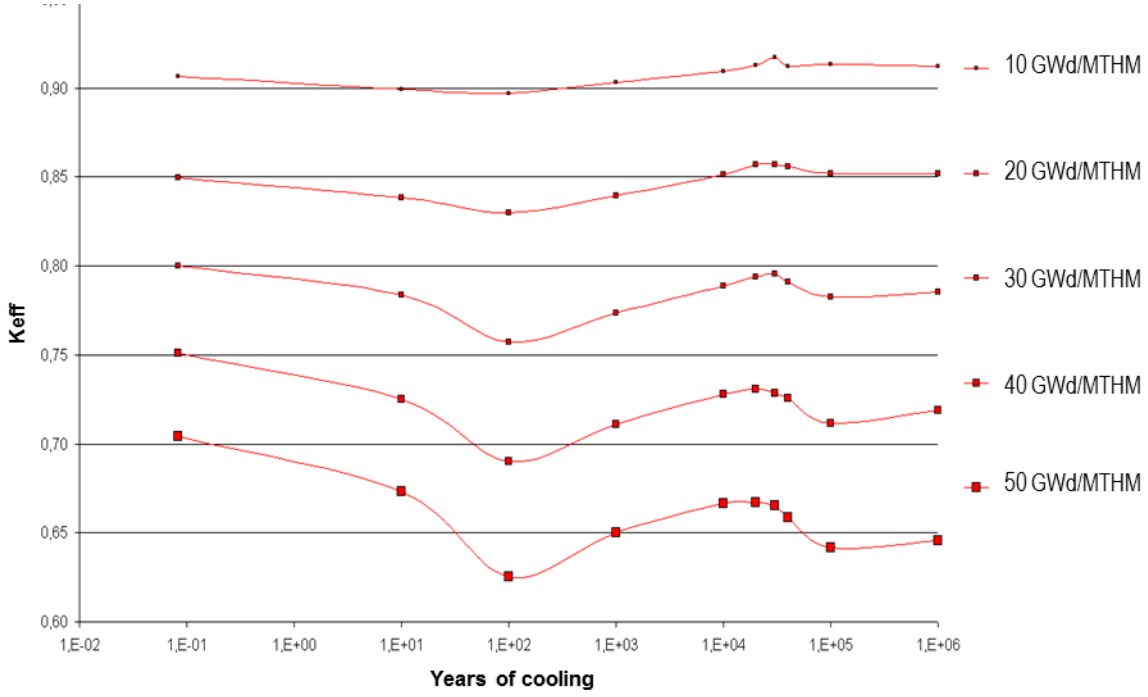


Figure 10: Changes in the k_{eff} value of a storage cask according to the cooling time and burnup

Essentially, the change in the reactivity of the used fuel assemblies without flux consists of three main phases.

Between leaving the reactor and around 100 years of cooling, the decay:

$$^{241}Pu \xrightarrow[T_{1/2}=14.3\,yrs]{} {}^{241}Am$$

causes a significant decrease in reactivity.

Between 100 years and around 30,000 years of cooling, the decay:

$$^{241}Am \xrightarrow[T_{1/2}=432\,yrs]{} {}^{237}Np$$

$$^{240}Pu \xrightarrow[T_{1/2}=6555\,yrs]{} {}^{236}U$$

causes a significant increase in reactivity.

Between 30,000 years and around 1 million years of cooling, the decay:

$$^{239}Pu \xrightarrow[T_{1/2}=24204\,yrs]{} {}^{235}U$$

causes a decrease in reactivity.

Depending on the nature of the problem, different cooling times will need to be considered:

- For handling or interim storage operations that start during the first few hours after reactor shutdown, possibly extending over a few decades, a fixed cooling period of 30 days must be taken into account to provide the most penalizing used fuel composition, particularly through the reaction:

$$^{239}Np \xrightarrow[T_{1/2}=2.35\,days]{} {}^{239}Pu$$

- For handling or interim storage operations that start x years after reactor shutdown (x >> 30 days), which may extend over y years (y < 100 years), a maximum cooling period of x years may be taken into account to take advantage of the fall in reactivity following an initial cooling phase.

The findings from this study are in line with the results obtained by Parks and Wagner in reference 18.

3.1.2 WORKING GROUP CONCLUSIONS ON THE IRRADIATION CONDITIONS

Initially, the Working Group was interested in PWR UOX fuel, with initial ^{235}U enrichment of up to 5 wt%, and a burnup less than or equal to 60 GWd/MTHM.

The conclusions below were drawn from the studies conducted at IRSN, EDF, CEA and SGN and, in particular, from reference documents 17, 21 and 30.

Effect of neutron spectrum hardening

It appears that the neutron spectrum can be modified by: the presence of control rods, the soluble boron concentration, the temperature of the moderator, the presence of burnable poison, or the presence of MOX fuel in the immediate vicinity of the UOX fuel.

The consideration of control rods during irradiation is conservative. Reference documents 8, 20, 21 and 30 show that, for an isolated assembly (with initial enrichment between 3 wt% and 5 wt%), the difference between taking account of the control rods and disregarding them leads to an increase in reactivity of up to 5,000 pcm (in Δk) for a burnup of 40 GWd/MTHM.

Reference document 21 also shows (see Appendix 10) that for an infinite lattice of PWR UO2 assemblies, initially enriched at 4.5 wt% and irradiated at 40 GWd/MTHM, the reactivity penalty (Δk) due to the full axial insertion of a control rod during a cycle is between 500 pcm (if the control rod is inserted during the first cycle) and 2,100 pcm (if it is inserted during the third cycle).

The investigations carried out by EDF on around 40,000 assemblies, in five major assembly families, made it possible to determine the proportion of assemblies that were irradiated in the presence of control rods, depending on the core management method (see Appendix 10). The findings from this study show that, regardless of the management method, fewer than 1% of the assemblies can be irradiated with the presence of control rods in all cycles, and around 50% of the assemblies are never irradiated with control rods inserted. It can also be seen that around 23% of the assemblies were irradiated with control rods inserted during their last irradiation cycle.

The table below shows the insertion history of the rods, depending on their function.

Table 4: Different control rods used in French reactors

Type of rod	Function	Insertion during irradiation
S	Emergency shutdown rod	Rods extracted (inserted by 5cm)
R	Control rod (temperature)	Variable (depending on the power level, control and management of the core) Represents around 10% of the fissile height at nominal power
N	Black rod (power)	Variable Fully extracted at nominal power
G	Grey rod (power)	Variable Fully extracted at nominal power

Maximizing the boron concentration maximizes the associated spectrum effect. The studies conducted at CEA showed that modelling the actual concentration variations during the cycle had little effect on the concentration of actinides (the difference observed when using the DARWIN package – see paragraph 3.2.2 - to validate PWR MOX fuel was less than 0.5% for the concentrations of uranium, plutonium, americium and neptunium - see reference 22. Thus, a calculation performed using a constant concentration value equal to the maximum boron concentration value gives a maximum k_{eff} value.

For the moderator temperature, it is conservative to consider the core maximum output temperature.

The presence of burnable poisons is also likely to lead to a hardening of the neutron spectrum. A study was conducted at CEA to assess the impact that UOX assemblies irradiated with gadolinium rods would have on the k_{inf} value. In France, some core management methods include fuel assembly reloads containing gadolinium rods.

These gadolinium rods may have a depleted uranium (0.715 wt%) or an enriched uranium (2.5 wt%) support. As an example, the CYCLADES management method for the PWR 900 CP0 series includes, out of the 52 reloaded assemblies, 36 assemblies with 12 gadolinium rods (8 wt% gadolinium) and uranium support equal to 0.715 wt%. HTC1 management (900 CPY PWR) includes, among the 48 reloaded assemblies, 32 assemblies with 12 gadolinium rods (8 wt% gadolinium) and 2.5 wt% uranium support, and 16 assemblies with 4 gadolinium rods (8 wt% gadolinium) and 2.5 wt% uranium support. The GEMMES, GALICE and HTC2 management systems for the 1300 PWR, and the ALCADE method for the 1450 PWR, also involve reloads with assemblies with gadolinium.

The study therefore involved performing a depletion calculation (using the APOLLO2 code) for an assembly containing a certain number of gadolinium rods, then recalculating k_{inf} using the CRISTAL package. The first part of the study showed that the presence of gadolinium rods had a low impact on the BUC nuclide concentration, with the exception of the plutonium and americium isotopes, whose concentrations at low burnup increased by a few percent. These effects diminished with the burnup, reflecting the consumption of the gadolinium (see Appendix 11).

In the case of an interim storage pond, the presence of UO_2-Gd_2O_3 rods reduced K_{inf} by 3,000 pcm at 2 GWd/MTHM, due to the presence of gadolinium 155 in the gadolinium rods. This decrease became less marked as soon as the burnup exceeded 20 GWd/MTHM. In the light of these results, it is therefore conservative to take account of the fact that the

assemblies were irradiated without gadolinium rods in the depletion calculation diagram for burnup credit.

A study by AREVA-NP showed that for an enrichment of gadolinium rods between 3 wt% and 4 wt%, the conclusions are not the same. For example, if the enrichment of rods containing gadolinium is equal to that of rods that do not contain gadolinium, an assembly with gadolinium that has undergone more than one irradiation cycle will be more reactive than an assembly that does not contain gadolinium rods with the same burnup.

Furthermore, the studies conducted for the LINGAO nuclear plant (presented in Appendix 12) revealed distortion in the axial burnup profile due to the presence of gadolinium rods. Thus, for a burnup of 37.7 GWd/MTHM and initial ^{235}U enrichment of 4.45 wt%, the presence of eight gadolinium rods during irradiation gives the profile a residual penalty of around 100 pcm for each gadolinium rod, due to the fact that the gadolinium was not used at the ends.

From the internal NRC document, reference 23, it appears that the burnable poisons used in the USA can be grouped into two categories:

- burnable poisons in the form of absorbent rods inserted into tubes (BPRs); this type of poison leads to a more reactive situation after irradiation than when the assembly does not contain burnable poison (in particular, the neutron spectrum is hardened by the presence of the absorbent rods, and when these rods are extracted, generally at the end of the first irradiation cycle, this results in a moderating ratio within the fuel that is more favorable to the chain reaction);
- burnable poisons in the "integral" form (IBAs); the absorbent is integrated into the cladding of certain fuel rods (IFBAs) or is mixed with uranium oxide; according to the above study, the impact of their presence on the effective multiplication factor remains low and, for assemblies that do not contain IFBAs, leads to a lower reactivity after irradiation than for assemblies irradiated without burnable poison (this decreasing reactivity is mainly linked to the insufficient quantity of fissile matter at the beginning of life).

With regard to the effect of the presence of MOX fuel during irradiation in close proximity to the UOX fuel, the following points have emerged from the calculations performed by CEA, which are presented in Appendix 2:

- the presence of four MOX assemblies placed at the four corners of the UOX assembly has no obvious effect on the concentration of the nuclides selected for our studies;
- the presence of four MOX assemblies, placed opposite the UOX assembly, slightly modifies the concentrations of ^{236}U, ^{238}U, ^{240}Pu, ^{103}Rh, ^{133}Cs and ^{143}Nd; however, this

configuration significantly modifies the concentrations of ^{235}U, ^{239}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am, ^{237}Np, ^{149}Sm, ^{151}Sm, ^{152}Sm and ^{155}Gd.

From the study presented in reference 21 and detailed in Appendix 2, it was found that the reactivity of a lattice of UOX assemblies in the pool increased when each UOX assembly was irradiated with a complete MOX environment. This increase ranged from 40 pcm for the assemblies irradiated up to 10 GWd/MTHM to 1,250 pcm for assemblies irradiated at 40 GWd/MTHM.

This latter case corresponds to a very high and unrealistic MOX usage rate, as it would occur during the entire irradiation process; the effects are therefore enhanced here.

Specific power

It is conservative to adopt the maximum specific power, provided that a particular process is used for the ^{149}Sm, ^{151}Sm and ^{155}Gd.

The findings presented in paragraph 3.1.1.3 show that the hypothesis of a maximum specific power will maximize the concentrations of the fissile nuclides and will minimize the concentrations of the "absorbent" nuclides.

However, taking ^{149}Sm and ^{151}Sm into account, with a maximum specific power, is not conservative in principle (since the concentration of these absorbent nuclides decreases when the specific power reduces). Similarly, the calculation of the concentration of ^{155}Gd, which depends mainly on the decay of ^{155}Eu, will not be conservative when the specific power is increased, given the overestimation of the production of ^{155}Eu. Finally, we should insist on the fact that this phenomenon will be particularly sensitive at the ends of the assembly, where the specific power during irradiation is the lowest. Studies are therefore required to determine correction factors for the concentrations of the four nuclides mentioned above; these must take account of the overestimation of the concentrations of these nuclides, combined with the hypothesis of an increased specific power.

Another possible approach would involve reasoning in terms of overall conservative used fuel composition rather than in terms of used fuel composition, nuclide by nuclide. The CEA and IRSN reports in references 21 and 30 show that reactivity has a low sensitivity to the specific power, particularly within the realistic range of 30-50 W/g. A bounding value could therefore involve selecting a specific power that is slightly greater than the mean specific power in the reactor.

Irradiation history

With regard to the effect of the possible presence of inter-cycle periods, and the effect of variations in the specific power during the cycles, the study cited in reference 14, together with a study conducted by the DOE in reference 18, indicates that the consequences of such variations on the effective multiplication factors are small[5] (about a few hundred pcm). Moreover, given the conservatisms used, and in order to simplify the calculations, it seems acceptable to retain only one irradiation cycle.

Fuel temperature

For the mean fuel temperature, it is more conservative to consider a temperature that is higher than the maximum mean temperature reached during normal operation.

Cooling time

The needs of the different members of the Working Group differ according to the nature of the studies carried out:

- for transportation and interim pool storage, minimum cooling times are greater than or equal to 6 months;
- for reprocessing, minimum cooling times are greater than or equal to 3 years;
- for pools holding fuels irradiated in the reactor, typical cooling times are between 0 days and 3 months;
- for in-containment fuel storage, minimum cooling times are greater than 30 or 50 years;
- for storage in deep geological layers, minimum cooling times are greater than 100 years.

Reactivity decreases over a cooling period of 100 years, and then increases over a period of around 30,000 years, all things being equal (same geometry, same environment, absence of corrosion, etc.).

[5] There is a connection between the irradiation history and the cooling time, which plays a part in determining the most penalizing situation. If we consider two irradiation histories, the first involving continuous irradiation without inter-cycles, and the second having six irradiation cycles with a two-year shutdown before the final cycle, it appears (according to the ORNL document, reference 18) that the concentrations of ^{155}Eu and ^{241}Pu after irradiation decrease in the second case, as well as the concentrations of ^{155}Gd and ^{241}Am. This is because, after irradiation, the ^{241}Pu and ^{155}Eu, which disappeared during the inter-cycle period, did not have the time to reach the values obtained before the inter-cycle. The ^{241}Am and ^{155}Gd produced during the inter-cycle period are consumed when irradiation resumes. Thus, for a cooling time of five years, the predominant effect will be the fall in the number of ^{241}Pu atoms: the most penalizing case consists of one continuous irradiation cycle. On the other hand, for a cooling time of 200,000 years (over ten times the decay period for ^{241}Pu and ^{241}Am), the predominant effect is the decrease in ^{155}Gd (linked to the decrease in ^{155}Eu) when an inter-cycle period occurs: the most penalizing case consists of the presence of an inter-cycle period at the end of irradiation. However, the differences in terms of Δk remain very small (less than 500 pcm).

Consequently, for operations or configurations in which the cooling time is less than 100 years, is seems that the minimum cooling time guaranteed by the operator may be used (for cooling periods of less than 20 days, consideration should be given to the fact that the concentration of ^{239}Pu is increased by the concentration of ^{239}Np[6]). However, when a study uses a non-zero cooling time, it is essential:

- either to ensure that the operation is specific for a "de facto" cooling time of less than 100 years (such as in the case of transportation);
- or, for interim storage operations where the cooling time CT_0 used in the criticality studies is not zero, to guarantee that the reactivity during storage never exceeds the level obtained during the cooling time CT_0 (see references 19 and 24 together with the study described in Appendix 9).

With regard to this last point, for operations or storage subject to a cooling time that could exceed 100 years, it would be possible to disregard the ^{241}Am and, if necessary, the ^{240}Pu, in the studies. The study presented in Appendix 9 appears to show that, in general, for handling or interim storage operations that take place z years after reactor shutdown (z < 1,000 years), a fixed cooling period of 30 days will guarantee the conservative nature of the calculation. However, if it can be demonstrated that there is a cooling period of between 30 days and 100 years that can guarantee the penalizing nature of the calculation, then this time may be used to reduce the calculated reactivity.

For handling or interim storage operations that take place t years after reactor shutdown (t > 1,000 years), an increase in reactivity over and above that recorded initially after 30 days of shutdown is possible but not systematic (see Appendix 9); the conservative nature of the hypotheses must therefore be demonstrated, given that the peak in reactivity occurs between 20,000 and 40,000 years.

3.2 CONCENTRATIONS OBTAINED FROM THE DEPLETION CODES

3.2.1 DEPLETION CALCULATION CODES

The calculation of the nuclide concentrations in the used fuel for a given burnup is commonly known as "depletion calculation". It involves resolving the depletion equations for each nuclide, taking account of the initial composition of the fuel, its irradiation history (burnup,

[6] The ^{239}Np transforms into ^{239}Pu over a period of around 2.5 days.

specific power, flux spectrum, cooling time, etc.), and nuclear data (cross-sections, decay constants, decay chains, etc.).

3.2.2 FRENCH CALCULATION CODES

The APOLLO, DARWIN and CESAR codes are the calculation codes used by the Working Group on burnup credit for the parametric studies. The studies may be conducted with either one of the depletion codes, in the knowledge that the concentrations obtained can be modified by a correction factor, which depends on the validation of the code and the source data used.

The DARWIN V2.0 code is fed by cross-section libraries (Saphyb) from APOLLO 2.5 calculations with the CEA 93.V6 library. The CESAR V5 code is also fed by cross-section libraries from APOLLO 2.5 calculations with the CEA 93.V6 library; this is a simplified code which can be used to interpolate tabulated results to determine the concentrations of a used fuel.

The DARWIN (PEPIN 2) and CESAR depletion modules resolve the fuel depletion equation using cross-sections collapsed into one group for all of the capture reactions. For decay, the CESAR and DARWIN codes use the values from the JEFF 2.2 evaluation.

The DARWIN code underwent a validation programme, presented in Appendix 3, which was based on determining the differences between the calculated concentrations and the measured concentrations ((C-E)/E):

- in the used fuel samples taken at the mid-point of the used fuel assemblies;
- in "dissolution juices" at La Hague, involving the dissolution of two assemblies at most.

The CESAR code was validated in relation to the DARWIN code.

With regard to this last point, it is important to note that the analyses conducted on "dissolution juices" (providing the mean concentration of a given nuclide $\overline{C(X)}$ for the mean burnup $\overline{BU}$) are only able to validate a "dissolution" approach. As the concentration of the nuclides is not proportional to the burnup, these analyses cannot validate the depletion calculation code for determining a concentration (C(X)) for a burnup BU (even if BU = $\overline{BU}$).

This basis for validation covers the validation of the used fuel composition for UOX fuels with enrichments of 3.1 wt% to 4.5 wt% for burnups of 10 GWd/MTHM to 60 GWd/MTHM. Reference document 25 shows that the results were satisfactory for nuclides ^{234}U, ^{235}U, ^{236}U, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{241}Am: (C/E)-1 $\leq$ 5%. Nuclides ^{238}Pu and ^{242}Pu are underestimated by -11% and -7% respectively. With regard to the burnup credit fission products, the validation results are satisfactory for neodymium isotopes 143 and 145, samarium isotopes 147, 150 and 152, and

cesium 133. Only the ^{151}Sm, ^{153}Eu and ^{155}Gd are significantly overestimated, with calculation/experiment differences of 13%, 16% and 12% respectively at 60 GWd/MTHM.

For metal fission products, the current analytical results do not provide a conclusion (due to an insolubility problem from 30 GWd/MTHM). However, only silver 109 is highly underestimated (see Appendix 5).

In order to take account of the burnup credit in criticality studies, correction factors to be applied to the concentrations of the selected nuclides are currently being defined for the CESAR and DARWIN depletion codes (see Appendix 7, documents 26 and 27, and articles 28 and 29).

In order to assess the effect of correction factors on the concentration of actinides, calculations performed by IRSN and reported in reference 30 demonstrated the effect of a 1% variation in the nuclide concentrations, for a fuel initially enriched at 4.5 wt% and for burnups lower than or equal to 44 GWd/MTHM. The main effects are shown in Table 5 below.

Table 5: Effects (determined from a perturbation calculation) of a 1% variation in the concentrations of U^{235}, Pu^{239}, Pu^{241} and Pu^{240} on the reactivity of a fuel element, with initial enrichment equal to 4.5 wt% and irradiated up to 44 GWd/MTHM, and for a non-perturbed (standard) burnup profile

Nuclide variation of +1%	**^{235}U**	**^{239}Pu**	**^{241}Pu**	**^{240}Pu**
Δk	+ 104 pcm	+ 75 pcm	+ 29 pcm	- 28 pcm

Finally, it should be noted that the DARWIN and CESAR depletion codes provide the final number of atoms for each nuclide per metric ton of heavy metal. However, in order to calculate the effective multiplication factor with CRISTAL, we need to know the concentration of the nuclides in atoms/cm^3. Also, it is conservative to use the density of the fresh fuel applied to the selected nuclides (which means artificially increasing their number by around 1%).

3.3 CONSERVATIVE CONCENTRATIONS

To conclude, in order to determine the nuclide concentrations using calculation codes such as APOLLO, CESAR or DARWIN, it is conservative to adopt the irradiation conditions defined in the previous paragraphs, namely:

- the presence of control rods;

- the presence of an environment of MOX fuel assemblies;
- a boron concentration greater than the mean concentration during the cycle;
- the output temperature of the core for the moderator;
- the maximum mean effective temperature of the fuel;
- the maximum specific power during a single irradiation cycle, possibly with, for certain nuclides (^{149}Sm, ^{151}Sm, ^{242}Pu and ^{155}Gd (child of ^{155}Eu)) the incorporation of correction factors. However, a study has shown that a specific power that is slightly higher than the mean power will not change the k_{eff} values significantly.

The values to be used with regard to the geometry of the used fuel remain to be defined.

To ensure conservatism when taking burnup credit into account, the concentration values obtained by the calculation will be modified by a correction factor, resulting from the validation of the version of the depletion calculation code used (APOLLO, CESAR or DARWIN).

Moreover, the density retained for the used fuel, consisting of only those nuclides selected for the approach, may, conservatively, be taken to be equal to the density of the fresh fuel.

Some of these constraints could be relaxed, as the associated demonstrations are yet to be developed in each specific case.

4 METHOD FOR CALCULATING THE EFFECTIVE MULTIPLICATION FACTOR

The previous chapters have listed the hypotheses that must be used to ensure that the characteristics of the modelled used fuel assembly are conservative, particularly with regard to the irradiation history, the burnup profile and the measurement methods for checking the burnup.

The used fuel assemblies, with their burnup profile and nuclide concentrations, are the input data for the criticality calculations, which are addressed in this chapter. The paragraphs that follow describe:

- the calculation methods used to obtain the multiplication factor, based on the burnup profile and the concentrations associated with each axial zone of the fuel;
- the validation of the calculation methods, particularly in view of the consideration of fission products;
- the problems associated with the calculation methods used (notably the Monte Carlo method), given the loose neutron couplings between the two ends of a used fuel assembly, together with the proposed solutions.

4.1 CALCULATION SEQUENCE

The calculation sequence described below is based on the CESAR and DARWIN depletion calculation codes, together with the CRISTAL criticality package. This description is only one example of a calculation sequence; other sequences based on the same principles may also be considered (such as STARBUCS using the ORIGEN code and the SCALE system).

The effective multiplication factor is determined using the CRISTAL V1 package, which was subject to the specifications in reference 31.

The nuclide concentrations, derived from a CESAR V5 or a DARWIN V2.0 calculation, are computed for each burnup value, corresponding to the different axially divided zones selected.

Concentration correction factors are introduced to take account of the calculation/experiment difference between the used fuel compositions, and also, where necessary, to correct their cross-sections and take account of the overestimation of certain absorbent nuclides when a maximum specific power is assumed.

The effective multiplication factor is calculated using the APOLLO and MORET codes ("standard" route) or the TRIPOLI code ("reference" route; in this case, the cross-sections are taken directly from the libraries used by TRIPOLI and the input data are the nuclide concentrations for each axial zone). The input files for these codes contain the geometric description of the modelled system with, for the used fuel assemblies, a number of zones with the zone heights defined in advance, based on the burnup profile selected for the study.

All of these stages are represented in Appendix 4 (for the CRISTAL V1 “standard” route).

4.2 VALIDATION OF THE CROSS SECTION LIBRARIES AND CRITICALITY CALCULATIONS

The validation of the cross-sections of the fission products used is based on:

- the oscillation experiments conducted in the MINERVE reactor in Cadarache;
- the "Fission Products" experiments conducted at the Apparatus B experimental facility in Valduc.

With regard to the oscillation experiments (see Appendix 6), the central rod oscillation spectrum is characteristic of "fuel transportation" and "pool storage" in the $R1UO_2$ configuration, and "fuel dissolution" in the $R2UO_2$ configuration; these experiments are able to determine the reactivity of a sample, with either an inert (alumina matrix) support or a uranium oxide support, doped with the examined fission product. By making a comparison with CRISTAL calculations, it is possible to determine whether the cross-section of the fission product studied, collapsed into one group on the core energy spectrum used during the experiment, is known in a satisfactory way. During the Burnup Credit programme associated with these experiments, 15 fission products[7] were tested.

The interpretation of the oscillation experiments on separate samples of fission products (see reference 32) updated the overestimation of ^{99}Tc and ^{133}Cs (4% in $R1UO_2$). The calculation/experiment discrepancies are satisfactory with regard to the ^{153}Eu, ^{109}Ag, ^{152}Sm, ^{145}Nd and ^{155}Gd (underestimation of less than 2%). The ^{143}Nd and ^{95}Mo were underestimated by 4% in $R1UO_2$, and the ^{149}Sm by 6%. For the ^{103}Rh, the interpretation of the oscillation experiments revealed a considerable overestimation (13% in $R1UO_2$), which would need to be moderated due to the fact that the dissolving technique induces fairly significant analytical uncertainties.

[7] The 15 fission products are those recommended by the OECD, i.e. the "first" six fission products (^{149}Sm, ^{103}Rh, ^{143}Nd, ^{133}Cs, ^{152}Sm and ^{155}Gd) together with the following nine fission products: ^{101}Ru, ^{95}Mo, ^{99}Tc, ^{147}Sm, ^{150}Sm, ^{151}Sm, ^{145}Nd, ^{109}Ag and ^{153}Eu.

No information or trend could be ascertained in relation to the individual poisoning of nuclides ^{147}Sm, ^{150}Sm, ^{151}Sm or ^{101}Ru, as the capture of these samples is largely attributable to the residual ^{149}Sm or the ^{99}Ru with regard to the ^{101}Ru.

Furthermore, it appears that to obtain the above results, the multi-group cross-sections of the fission products have to be self-shielded.

As for the "Fission Products" experiments conducted at the Apparatus B experimental facility in Valduc from 1997 to 2004 on installation B, described in references 33, 34, 35, 36 and 37, these involve the "first" six fission products[8] and are divided into three different configuration types:

- the "physical" type experiments: a tank filled with a solution containing a given fission product (or a mixture of fission products) is placed in the center of a control lattice of uranium oxide rods; these experiments are used to validate the cross-sections of each separate fission product in spectra similar to those encountered during transportation and in dissolution solutions;
- the "elementary dissolution" type experiments: a tank filled with a solution containing either a given fission product, surrounded by a control lattice of UOX or HTC fuel rods[9], or a mixture of fission products and uranyl nitrate, with a lattice of UOX or HTC fuel rods, is placed in the center of a lattice of uranium oxide rods; these experiments are used to validate the interactions between the cross-sections of the fission product and the actinides and, therefore, to validate the self-shielding models (resonance overlap).
- the "global dissolution" type experiments: a tank filled with a lattice of UOX[10] or HTC fuel rods is placed in a solution of depleted uranyl nitrate containing a mixture of the six fission products; these conditions are very similar to the environment encountered in a dissolver, with no self-sustained critical reactions in the system.

The experiments were conducted then modelled and used with the CRISTAL package. The following comments can be made on the results obtained:

- in the experiments, the reactivity worth of each fission product (negative reactivity provided by the fission product) is high enough to be able to validate its absorption

[8] The ^{95}Mo will also be used in the FP experiments described in reference 33.

[9] The HTC fuel rods (high burnup fuel rods) were manufactured to contain the quantity of uranium and plutonium corresponding to uranium oxide initially enriched at 4.5 wt% and irradiated to reach a burnup of 37,500 MWd/MTHM.

[10] In the configuration with UOX rods, the solution does not contain any FPs.

(around 4,000 pcm); only the ^{133}Cs and ^{143}Nd experience a slightly lower negative reactivity due to their smaller quantities in the solution;

- from 100 to 400 pcm, the effective multiplication factors obtained by the calculation (using the CRISTAL V1 "standard" route) are still[11] slightly higher than 1, which is conservative;
- the propagation of the experimental uncertainty in terms of Δk_{eff} is less than 100 pcm (1 σ).

In addition, the calculations performed in the normal way using the CRISTAL package (the identification details of which are provided in reference 38) are validated based on the method described in reference document 39.

4.3 CALCULATION CONVERGENCE AND LOOSE NEUTRON COUPLINGS FOR CODES USING A MONTE CARLO METHOD

In the used fuel storage or transport configurations, most of the reactivity comes from the ends of the fuel assemblies, which are the least irradiated parts. The central part of the assembly, separating the two ends, is particularly large compared to the mean free path of the neutrons. Furthermore, this zone contains fission products resulting from irradiation, which are highly absorbing. These two factors (distance from the fissile zones and separation by the absorbing materials) mean that the two ends of an assembly are loosely coupled from a neutron point of view. In other words, the probability of a neutron passing from one fissile zone to another is relatively low.

With any of the criticality calculation codes using the Monte Carlo method, converging towards the correct effective multiplication factor value can be problematic when the configuration contains loosely coupled fissile units. This is because a Monte Carlo calculation is iterative, and the source sites of the neutrons simulated during one step are chosen from the fission sites stored during the previous step. The calculation bias may thus result from the incorrect initial positioning of the neutrons by the user, an insufficient number of neutrons simulated at each step of the calculation, or an insufficient number of calculation steps, culminating in an under-representation, or even the omission, of particularly reactive fissile zones.

After a calculation has been performed, the fuel zones at the ends of the assemblies must be checked to ensure that they are "visited" by a sufficient number of neutrons.

[11] With the exception of the ^{103}Rh (elementary dissolution) for which a slight underestimation of around 300 pcm is observed.

To counter this "source convergence" problem inherent to using the Monte Carlo method for criticality calculations, different methods for simulating and sampling the source neutrons from the fission sites have been refined by the teams developing the Monte Carlo computer codes, and have been subsequently tested.

The source convergence problem is being studied by the OECD / NEA "Source Convergence" Working Group, and the MORET 4 code in the CRISTAL package is being developed, to make it easier to position the sources and detect poor calculation convergence.

Four simulation methods, which are described in Appendix 13 and presented in reference 40, are available in the MORET 4 code. The effectiveness of each one is currently being examined for the "Source Convergence" Working Group.

To describe the distribution of the source neutrons in the first generation of a Monte Carlo calculation, it is currently recommended to proceed as follows:

- use the SUNI option (the source neutrons are distributed in equal number across all of the fissile volumes), which guarantees that all of the fissile volumes will be "visited" during the first generation, and thus reduces the likelihood of a fissile volume being "forgotten" throughout the simulation;
- place additional source neutrons at the ends of the assemblies to more accurately reproduce the supposed spatial distribution of the neutrons.

It is evident that the standard checks, relating to the convergence curve profile and the variations in the final multiplication factor value if the first steps in the calculation are removed, are still required.

Finally, it should be noted that, when the fuel is highly irradiated and most of the reactivity comes from the ends of the used fuel assembly, the initial placement of the sources in this zone leads to an overestimation of the effective multiplication factor in the first steps in the calculation (the spatial distribution of the neutrons does not, therefore, take complete account of the neutron leakage outside these volumes) and, consequently, an overestimation of the final effective multiplication factor. This choice is therefore conservative.

4.4 CONCLUSIONS ON THE CALCULATION METHODS

Taking burnup credit into account complicates the criticality calculations and therefore increases the risk of modelling errors. This problem spurs the need to develop appropriate computing sequences to automatically take account of burnup credit in the calculations.

The Working Group believes that particular attention should be paid to monitoring the data and assuring the quality of the computing sequences.

By way of example, the new CRISTAL V1 criticality package, for performing automated criticality calculations that take burnup credit into account, has been described.

Depletion calculations are first performed to determine the conservative concentrations of the different nuclides, to which correction factors may be applied, for each elementary axial zone linked to the discretization of the burnup profile. The k_{eff} calculation is then performed using the CRISTAL package.

The CIGALES code, a component of the CRISTAL V1 interface, is currently coupled with the CESAR 5 depletion calculation code. A similar coupling with the DARWIN V2.0 package is also available. The CIGALES code is designed to automatically generate APOLLO2 data files for the criticality calculations. It allows the user to select and take account of:

- correction factors (applied to the actinide and fission product concentrations), which are determined from the validation of the depletion calculations and the cross-sections;
- the assembly axial burnup profile;
- the number of axial zones modelled.

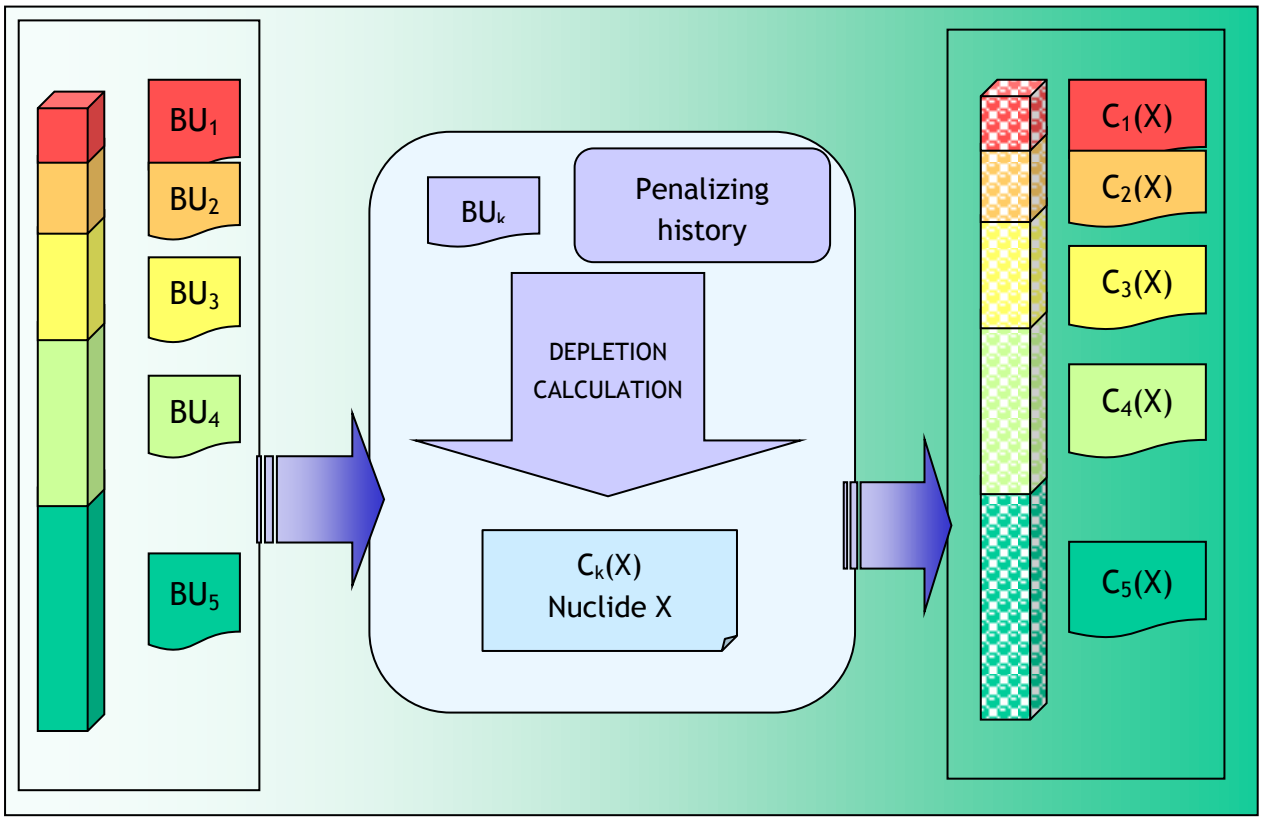
The links between the different calculations are shown in the two diagrams below (taken from Appendix 4).

The validity of the CRISTAL calculations depends on the validation of the cross-sections of the nuclides taken into account, together with the computer codes for the modelled configurations. In this regard, the experiments conducted in the MINERVE reactor in Cadarache, and at the Apparatus B experimental facility in Valduc, will enable us to determine whether it is necessary to apply correction factors to the concentrations and/or to take account of a calculation bias.

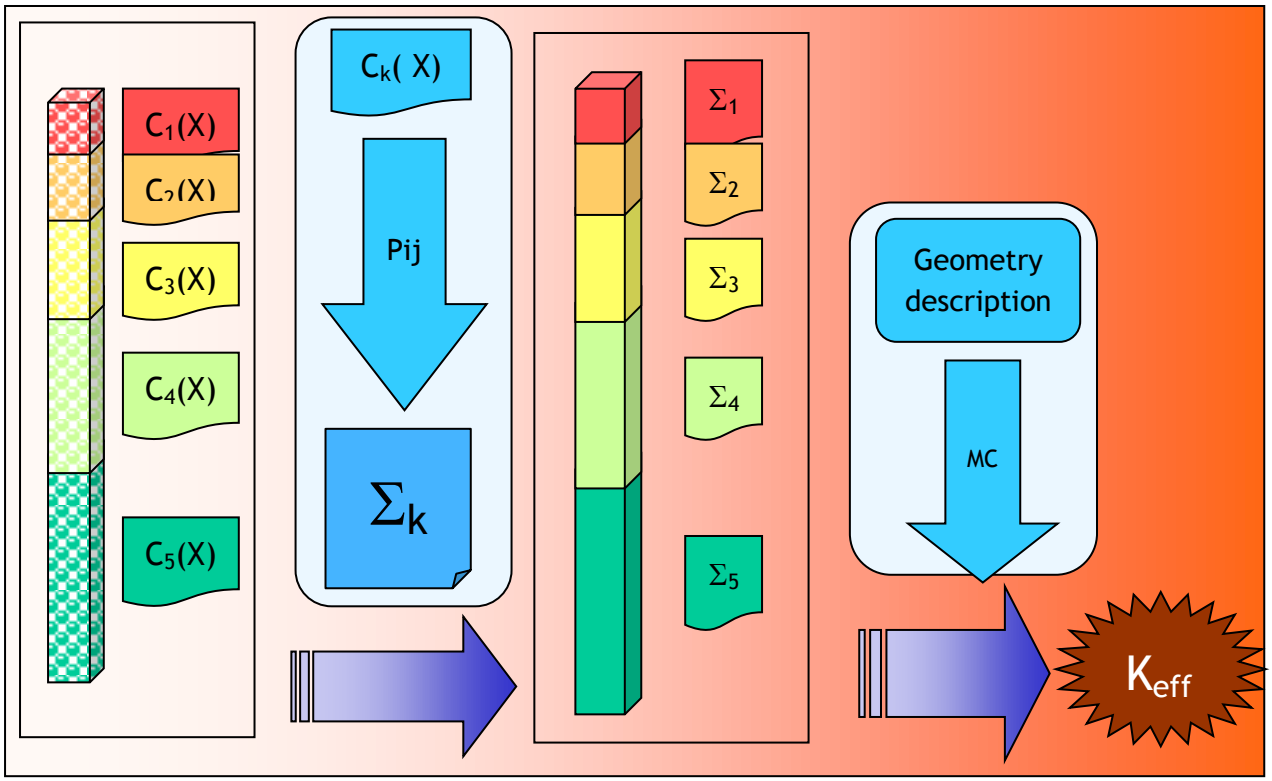
These calculations (taking fission products into account) are currently undergoing validation for the package “standard” route (APOLLO2-MORET4).

Finally, the modelled configurations involve assemblies whose reactivity is very highly dependent on the under-irradiation at the ends; but the ends of the fuel assemblies have very loose neutron couplings, which remains difficult to process in the Monte Carlo calculations. Consequently, certain precautions must be taken, particularly when initially positioning the neutron sources.

Depletion calculations performed without the CRISTAL package, based on a burnup profile (with conservatism verified)



Criticality calculations performed with the CRISTAL package, based on data provided by the depletion codes and modified by correction factors where necessary



5 CONCLUSIONS

The previous chapters have presented the various steps implemented in a criticality calculation that takes burnup credit into account. Furthermore, subject to the availability of information, sensitivity to the parameters involved in each of these steps has also been discussed. This report covers the domain of PWR UOX fuel with ^{235}U enrichment of less than 5 wt% and a burnup of less than 60 GWd/MTHM.

This approach suggests the following stages:

- determine the burnup profiles;
- discretize the previously selected profiles into N zones with a given burnup;
- for each of the axial zones, compute the concentrations of the selected nuclides using depletion codes. The first approach studied by the Working Group involved guaranteeing the conservatism of each stage; these calculations are performed with a penalizing irradiation history;
- apply correction factors to the concentrations of certain nuclides, bearing in mind the calculation/experiment discrepancies relating to the used fuel composition, and if necessary, the cross-sections and the variation in certain nuclides depending on the selected irradiation parameters;
- calculate the effective multiplication factor, adopting an additional margin for the criteria if necessary, depending on the validation being carried out.

The validity of the method is based on the validation of the burnup profile used for the calculations. The only validation method examined to date by the Working Group involved measuring the actual burnup and the axial burnup profile of the assemblies actually transported, handled or stored.

In certain cases, where the licensee does not wish to take any measurements, the use of a penalizing profile may be considered. However, the conservatism of the choice of profile must be provided by other profile validation methods. These will be presented in a revision of this document.

Each of these stages is dependent on modelling decisions, some of which have already been evaluated; the most conservative choices are shown in the table below.

	Conservative hypothesis
IRRADIATION HISTORY	Presence of a control rod
	Boron concentration greater than the mean concentration during the cycle
	Moderator temperature equal to the core maximum output temperature
	Fuel temperature equal to the maximum mean effective temperature
	Presence of MOX in the environment
	Irradiation during one single irradiation cycle
	Maximum specific power, with the possible use of correction factors for ^{149}Sm, ^{151}Sm, ^{155}Gd and ^{242}Pu or specific power slightly greater than the mean specific power
	For cooling times of less than 100 years, the cooling time is taken to be equal to the minimum guaranteed cooling time
PROFILES	Selection of a conservative profile
	Determination of the optimal number of axially divided zones
	Axial discretization, selecting the BU_{min} for each zone
	Maximum horizontal burnup gradient
MEASURE-MENTS	Incorporation of measurement uncertainties
KEFF CALCUL-ATION	Correction factors for the concentrations, from the validation of the used fuel composition calculation and the cross-sections of the fission products
	Fuel density with the selected nuclides equal to that of the fresh fuel
	Calculations with the Monte Carlo codes, initially placing sources in all of the fissile volumes and additional sources at the ends

Finally, the conservative approaches studied by the Working Group were found to have an impact on reactivity compared to the "best estimate" scenario, as shown in the table below:

	Calculation options	Effect on k_{eff} (1)
IRRADIATION HISTORY	Presence of a control rod during irradiation	At most: + 5,000 pcm
	Boron concentration equal to 600 ppm (instead of 456 ppm)	+ 300 pcm approx. (for a lattice of assemblies in the pool with a flat profile)
	Moderator temperature equal to the core maximum output temperature (330°C instead of 300°C)	+ 850 pcm
	Fuel temperature equal to the maximum mean effective temperature (650°C instead of 590°C)	+ 150 pcm
	Presence of MOX fuel in the environment	At most: + 1,200 pcm
	Irradiation during one irradiation cycle (without inter-cycle)	variation of +/- 500 pcm
	Maximum specific power equal to 60 W/g (instead of 30 W/g)	+ 450 pcm
	For cooling times of less than 100 years, the cooling time is taken to be equal to the minimum guaranteed cooling time	Dependent on the actual cooling time, between 0 and 7,000 pcm
	Cooling time greater than 100 years	Not quantified (see Appendix 9)
	Fuel density with the selected nuclides equal to the density of the fresh fuel	Not determined
	Presence of burnable poisons	Not determined (see Appendices 11 and 12)
	Geometry of the used fuel	Not determined (ACTION REQUIRED)
PROFILES	Determination of the penalizing profile from the series of profiles observed at La Hague	Not determined (ACTION IN PROGRESS)
	Use of a penalizing axial burnup profile (due to the insertion of the control rods throughout the irradiation process)	Up to + 12,000 pcm
	Use of a penalizing horizontal burnup gradient	< 500 pcm Up to + 1,600 pcm for 4 assemblies placed side by side
MEASURE-MENTS	Burnup measurement uncertainties	Not quantified (see Appendix 14)
FISSION PRODUCTS	Other fission products not taken into account	+ 3,000 pcm
KEFF CALCUL-ATION	Application of correction factors proposed by CEA (used fuel composition + cross-sections)	+ 600 pcm (for a lattice of assemblies in the pool, taking a burnup profile into account)

(1) Effects expressed in Δk for an isolated assembly initially enriched at 4.5 wt% and irradiated at 44 GWd/MTHM, as a result of taking account of the recommendation corresponding to a calculation, without taking the recommendation into account.

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17. "Effet de la prise en compte du taux de combustion pour des situations de stockage et de transport" ("Effect of taking the burnup into account for storage and transport conditions") - Report SEC/T/01.093 dated 8 March 2001 – French version only

18. "Parametric Analysis of PWR Spent Fuel Depletion Parameters for Long Term Disposal Criticality Safety" Oak Ridge National Laboratory - ORNL/TM-1999/99, August 1999

19. "Effet du burn up et du profil d'irradiation sur la réactivité à long terme de combustible irradié UOX" ("Effect of burnup and the burnup profile on the long-term reactivity of irradiated UOX fuel") EDF/SEPTEN report - ENPRNC040255, issue A – French version only

20. "A New Method to Take Burn-up into Account in Criticality Studies Considering an Axial Profile of Burn-up Plus Some Fission Products" - C. Lavarenne et al. - Proceedings of the ANS 2001 Winter meeting

21. CEA document

22. CEA document

23. Internal NRC document W6479/IIR-01-01

24. Compte rendu de réunion du groupe de travail sur le Crédit Burnup du 18 octobre 2004 (Minutes of the meeting of the Burnup Credit Working Group on 18 October 2004) – Document DSU/SEC/A/2005-104 dated 23 March 2005 – French version only

25. CEA document

26. CEA document

27. Compte rendu de réunion du groupe de travail sur le Crédit Burnup du 24 avril 2007 (Minutes of the meeting of the Burnup Credit Working Group on 24 April 2007) – Document DSU/SEC/A/2007-217 dated 19 July 2007 – French version only

28. "Recent advances in French validation program and derivation of the acceptance criteria for UOx fuel" – A. Barreau et al. - Proc. Technical meeting on Advances in Applications of Burnup Credit to Enhance Spent Fuel Transportation, Storage, Reprocessing and Disposition, London, August Sept 2005

29. "Determination of correction factors for the isotopic composition in Burnup Credit applications taking into account the depletion codes validation" – J. Raby et al. – Proc. Int. Conf. on Nuclear Criticality Safety, ICNC2007, St Petersburg, Russia, May 2007

30. "Etudes de différentes méthodes de calculs relatives à la prise en compte du taux de combustion pour des situations de stockage et de transport" ("Studies of different calculation methods for taking the burnup into account for storage and transport conditions") - Report SEC/T/02.018 dated 4 February 2002 – French version only

31. "Cahier des charges du formulaire CRISTAL Version V1" ("Specifications for the CRISTAL Version V1 package") SEC/T/01.035 dated 2 February 2001 – French version only

32. CEA document

33. "Exploitation du programme expérimental Produits de Fission – 1ère partie" ("Use of the fission products experiment programme – Part 1") Document DSU/SEC/T/2004-162/DR Issue A dated 17 November 2004 – French version only

34. "Exploitation du programme expérimental Produits de Fission – 2ème partie " ("Use of the fission products experiment programme – Part 2") - Document DSU/SEC/T/2004-441/DR Issue A dated 2 September 2005 – French version only

35. "Exploitation du programme expérimental Produits de Fission – 2ème partie " ("Use of the fission products experiment programme – Part 2") - Document DSU/SEC/T/2004-491/DR Issue A dated 2 September 2005 – French version only

36. "Exploitation du programme expérimental Produits de Fission – 3ème partie " ("Use of the fission products experiment programme – Part 3") - Document DSU/SEC/T/2004-504/DR Issue A dated 30 August 2005 – French version only

37. "Exploitation du programme expérimental Produits de Fission – 3ème partie " ("Use of the fission products experiment programme – Part 3") - Document DSU/SEC/T/2004-527/DR Issue A dated 16 September 2005 – French version only

38. "Notice d'identification de la version 1.1 du formulaire CRISTAL" ("Identification details for version 1.1 of the CRISTAL package") - Document DSU/SEC/T/2006-200 Issue A dated 16 April 2007 – CRISTAL-V1/NdI_V1.1/A – French version only

39. "Contribution à la qualification de la voie standard APOLLO2-MORET 4 du formulaire de criticité CRISTAL - Synthèse des résultats de calcul obtenus avec la version V1.1 du formulaire CRISTAL" ("Contribution to the validation of the APOLLO2-MORET 4 "standard" route in the CRISTAL criticality package - Summary of the calculation results obtained with version V1.1 of the CRISTAL package") - Document DSU/SEC/T/2007-1/D.R. – Issue A dated 26 February 2007 – French version only

40. "Development and comparison of Monte Carlo techniques implemented in the MORET4 code for the calculation of loosely coupled systems" Conference MC (Monte Carlo) 2000 - Lisbon 23-26 October 2000

7 APPENDICES

APPENDIX 1
DOCUMENT REFERENCE SYSTEM

This document reference system is structured around seven topics:

1- General method for taking account of the burnup

2- Used fuel composition and cross-section validation

3- Cross-section validation

4- Burnup profiles

5- Irradiation and cooling history

6- Discretization of axial burnup profiles and calculation methods

7- Burnup credit sequence validation

The various documents relating to each of these topics are listed in the remainder of this appendix.

1 General method for taking account of the burnup

1.1 Compte rendu de la réunion du 18 mars 1998 du groupe de travail sur le Crédit Burn-up (Minutes of the meeting of the Burnup Credit Working Group on 18 March 1998) SEC/A/99.218 dated 24 April 1998 – French version only

1.2 Compte rendu de la réunion du 3 mai 1999 du groupe de travail sur le Crédit Burn-up (Minutes of the meeting of the Burnup Credit Working Group on 3 May 1999) SEC/A/99.218 dated 1 July 1999 – French version only

1.3 Compte rendu de la réunion du 19 janvier 2000 du groupe de travail sur le Crédit Burn-up (Minutes of the meeting of the Burnup Credit Working Group on 19 January 2000) SEC/A/00092 dated 13 March 2000 – French version only

1.4 Actinide only Burn-up credit
DOE/RW-0472 Rev2, September 1998

1.5 Etude d'une méthode de prise en compte dans les études de criticité de la combustion massique des combustibles à base d'oxyde d'uranium des assemblages de type REP, en considérant un profil axial d'irradiation, l'évolution des actinides dans le combustible, ainsi que la formation des produits de fission (Study of a method for taking account of the burnup of uranium oxide-based fuels in PWR type assemblies in criticality studies, considering an axial burnup profile, actinide depletion in the fuel, and the formation of fission products)
SEC/T/02.005/DR dated 4 February 2002 – French version only

2 Used fuel composition and cross-section validation

2.1 CEA document

2.2 CEA document

2.3 CEA document

2.4 CEA document

3 Cross-section validation

3.1 CEA document

3.2 CEA document

3.3 CEA document

3.4 CEA document

3.5 CEA document

3.6 CEA document

3.7 Programme expérimental de type physique - Evaluation des expériences critiques relatives à la qualification des sections efficaces d'absorption de 6 PF (^{103}Rh, ^{133}Cs, ^{155}Gd, ^{152}Sm, ^{143}Nd) en solutions nitriques légèrement acides, seuls ou mélangés, dans une cuve de Zircaloy au centre d'un réseau pilote de crayons REP UO_2 à 4,738 % (poids) ^{235}U modéré et réfléchi par de l'eau (Physical type experimental programme - Evaluation of the critical experiments relating to the validation of the absorber cross-sections of six FPs (^{103}Rh, ^{133}Cs, ^{155}Gd, ^{152}Sm and ^{143}Nd) in slightly nitric acid solutions, alone or mixed, in a Zircaloy tank in the center of a control lattice of PWR UO_2 rods with 4.738% (weight) ^{235}U, moderated and reflected by water)
Document SEC/T/01.01/C.CEA dated 1 January 2001 – French version only

3.8 French Fission Products Burnup Experiments performed in Cadarache and Valduc. Results Comparison - J. Anno et al – in Proceedings of Int. Conf. on Nuclear Criticality Safety, ICNC'2003, Tokai Mura, Japan, 20-24 October 2003

3.9 Exploitation du programme expérimental Produits de Fission – 1ère partie – « *Expériences de type "Physique" relatives à la qualification des sections efficaces d'absorption de 6 Produits de Fission (^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{152}Sm, ^{155}Gd) en solutions nitriques légèrement acides, seuls ou mélangés, dans une cuve de zircaloy disposée au centre d'un réseau pilote, de crayons REP UO_2 enrichis à 4,738 % (poids) en ^{235}U, modéré et réfléchi par de l'eau* » (Use of the fission products experiment programme – Part 1 – *"Physical" type experiments relating to the validation of the absorber cross-sections of six fission products (^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{152}Sm and ^{155}Gd) in slightly nitric acid solutions, alone or mixed, in a Zircaloy tank in the center of a control lattice of PWR UO_2 rods enriched with 4.738% (weight) ^{235}U, moderated and reflected by water)*
Report DSU/SEC/T/2004-162/DR Issue A dated 17 November 2004 – French version only

3.10 Exploitation du programme expérimental Produits de Fission – 2ème partie – « *Expériences de criticité de type "Dissolution Elémentaire" : réseaux de crayons REP dans des solutions légèrement acides empoisonnées ou non par des Produits de Fission* » (Use of the fission products experiment programme – Part 2 – *"Elementary dissolution" type criticality experiments: lattices of PWR rods in slightly acidic solutions, with or without fission product poisoning)*
Report DSU/SEC/T/2004-441/DR Issue A dated 2 September 2005 – French version only

3.11 Exploitation du programme expérimental Produits de Fission – 2ème partie - « *Expériences de criticité de type "Dissolution Elémentaire" : réseaux de crayons REP immergés dans des solutions de gadolinium naturel, de nitrate d'uranyle appauvri empoisonnées ou non par des Produits de Fission (PF)* » *(*Use of the fission products experiment programme – Part 2 – *"Elementary dissolution" type criticality experiments: lattices of PWR rods immersed in solutions of natural gadolinium, of depleted uranyl nitrate with or without fission product poisoning)*
Report DSU/SEC/T/2004-491/DR Issue A dated 2 September 2005 – French version only

3.12 Exploitation du programme expérimental Produits de Fission – 3ème partie – « *Expériences de criticité de type "Dissolution dans une grande cuve non ajustée" sur des solutions de nitrate d'uranyle appauvri baignant un réseau de crayons UO_2* » *(*Use of the fission products experiment programme – Part 3 – *"Dissolution in a large non-adjusted tank" type criticality experiments on depleted uranyl nitrate solutions soaking a lattice of UO_2 rods)*
Report DSU/SEC/T/2004-504/DR Issue A dated 30 August 2005 – French version only

3.13 Exploitation du programme expérimental Produits de Fission – 3ème partie – « *Expériences de criticité de type "Dissolution avancée" : réseaux de crayons HTC dans des solutions de nitrate d'uranyle appauvri empoisonnées ou non par des produits de fission* » (Use of the fission products experiment programme – Part 3 – *"Advanced dissolution" type criticality experiments: lattices of high burnup rods in depleted uranyl nitrate solutions with or without fission product poisoning)*
Report DSU/SEC/T/2004-527/DR Issue A dated 16 September 2005 – French version only

4 Burnup profiles

4.1 Etude avec prise en compte du crédit Burn-up - Mise au point d'une méthodologie de calcul (Study taking account of burnup credit - Development of a calculation methodology)
CNAM engineering thesis presented by Mr Maillot and defended on 5 July 1999 – French version only

4.2 Effet de la prise en compte du taux de combustion pour des situations de stockage et de transport (Effect of taking the burnup into account for storage and transport conditions)
Report SEC/T/01.093 dated 15 May 2001 – French version only

5 Irradiation and cooling history

5.1 Proposition d'un protocole de calcul pour les études de criticité du combustible usé (Proposed calculation protocol for used fuel criticality studies)
Report EDF/DER/SNRE/DPR HT-11/99/021/A, October 1999 – French version only

5.2 Effet de la prise en compte du taux de combustion pour des situations de stockage et de transport (Effect of taking the burnup into account for storage and transport conditions)
Report SEC/T/01.093 dated 8 March 2001 – French version only

5.3 Etude de différentes méthodes de calcul relatives à la prise en compte du taux de combustion pour des situations de stockage et de transport (Study of different calculation methods for taking the burnup into account for storage and transport conditions)
Report SEC/T/02.018 dated 11 February 2002 – French version only

5.4 CEA document

5.5 Etudes de sensibilité des paramètres d'irradiation pour la prise en compte du Crédit Burnup dans les réacteurs REP-UOX (Effects of burnup and the burnup profile on the long-term reactivity of irradiated UOX fuel)
Report EDF/SEPTEN/ENPRNC040255 Issue A dated 2 June 2005 – French version only

6 Discretization of axial burnup profiles and calculation methods

6.1 Une contribution à la prise en compte du taux de combustion dans les études de criticité. Prise en compte de profil axial de combustion des assemblages combustibles (A contribution to the consideration of the burnup in criticality studies. Taking account of the axial burnup profile of fuel assemblies).
Report by Mr Tournade following the course held at SGN, March to July 1998 – French version only

6.2 Prise en compte du taux de combustion dans les études de criticité (Taking account of the burnup in criticality studies).
INSTN report by Mr Reverdy following the course held at SGN in 1999 – French version only

7 Burnup credit sequence validation

7.1 CEA document

7.2 CEA document

APPENDIX 2
SENSITIVITY STUDY ON THE EFFECT OF THE ENVIRONMENT

The purpose of this study is to examine the impact of UOX irradiation with a MOX environment on a criticality calculation. To do this, depletion calculations are performed using the DARWIN V2.0 fuel cycle package, followed by a criticality calculation on a pool storage configuration using the CRISTAL V1 criticality safety package (the chosen criticality configuration maximized the effects under examination).

To this end, three configurations are modelled in the depletion calculation:

1. Central UOX assembly irradiated without control rods, surrounded by eight UOX assemblies irradiated without control rods (serving as a reference);
2. Central UOX assembly irradiated without control rods, surrounded by eight MOX assemblies irradiated without control rods (to study a MOX environment irradiated without control rods);
3. Central UOX assembly irradiated without control rods, surrounded by eight MOX assemblies irradiated with control rods (to investigate a MOX environment irradiated with control rods).

In this study, the used fuel inventory used in the criticality calculations consists of the nuclides selected for the burnup credit studies, as follows:

- actinides ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{237}Np, ^{241}Am and ^{243}Am;
- the 15 fission products: ^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{152}Sm, ^{155}Gd, ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{109}Ag, ^{145}Nd, ^{147}Sm, ^{150}Sm, ^{151}Sm and ^{153}Eu;
- with a cooling time equal to zero.

In order to differentiate the used fuel composition problem from the "end effect" problem, the axial burnup profile is considered uniform and equal to the assembly mean burnup.

In the tables below, $\Delta\rho$ is the difference in reactivity defined by the formula:

$$\Delta\rho = \rho\,(\text{case2}) - \rho\,(\text{case1}) = Ln\frac{K_{eff}\,(\text{case2})}{K_{eff}\,(\text{case1})}$$

1. Comparison between the impact of a MOX environment and a UOX environment

The presence of MOX around a UOX assembly hardened the neutron spectrum, leading to an increase in the production of plutonium and, therefore, an increase in the reactivity of the UOX assembly lattice. This reached 1,250 pcm for a lattice of assemblies irradiated at 40 GWd/MTHM, which were completely surrounded by MOX throughout the entire irradiation process (see Table 6).

Table 6: Comparison between the effects of irradiation with a MOX environment and a UOX environment on the reactivity of a UOX lattice

BU of UOX assembly lattice	ρ (MOX env) - ρ (UOX env)
10 GWd/MTHM	+ 44 pcm
20 GWd/MTHM	+ 300 pcm
30 GWd/MTHM	+ 672 pcm
40 GWd/MTHM	+ 1,251 pcm

2. Influence of the MOX environment burnup

The impact of an irradiation with a MOX environment at 10, 20 or 30 GWd/MTHM compared to an irradiation with a MOX environment at 15 GWd/MTHM is shown in Table 7.

Table 7: Impact of a MOX environment burnup on the reactivity of a UOX lattice

BU of UOX assembly lattice	ρ (MOX $env_{10GWd/MTHM}$) - ρ (MOX $env_{15GWd/MTHM}$)	ρ (MOX $env_{20GWd/MTHM}$) - ρ (MOX $env_{15GWd/MTHM}$)	ρ (MOX $env_{30GWd/MTHM}$) - ρ (MOX $env_{15GWd/MTHM}$)
20 GWd/MTHM	- 1 pcm	- 23 pcm	- 50 pcm
40 GWd/MTHM	+ 83 pcm	- 117 pcm	- 257 pcm

It can be seen that the reactivity of the UOX lattice surrounded by a MOX environment at 30 GWd/MTHM during the irradiation, is lower than the reactivity of the lattice surrounded by a MOX environment at 15 GWd/MTHM during the irradiation. This is due to the decrease in the content of the ^{239}Pu in the MOX environment with the increase in its burnup, which softened the neutron spectrum of the UOX assembly.

However, the MOX environment burnup has a relatively low impact on the reactivity of the UOX fuel lattice.

3. Impact of a MOX environment irradiated with control rods

The MOX assemblies surrounding the UOX assembly are irradiated with inserted control rods throughout the irradiation process (at up to 15 GWd/MTHM).

The impact of an irradiation with a MOX environment irradiated with and without inserted control rods is shown in Table 8.

Table 8: Effects on the reactivity of a UOX lattice of an irradiation with a MOX environment irradiated with and without inserted control rods

BU of UOX assembly lattice	ρ (MOX $env_{with\ CRs}$) - ρ (MOX $env_{without\ CRs}$)
10 GWd/MTHM	- 3 pcm
20 GWd/MTHM	+ 3 pcm
30 GWd/MTHM	+ 37 pcm
40 GWd/MTHM	+ 30 pcm

Surrounding the UOX assembly with a MOX environment irradiated with inserted control rods leads to a hardened neutron spectrum, and thus to an increase of the reactivity of the UOX lattice. However, this effect is relatively small compared to the effect on the reactivity of the MOX environment itself.

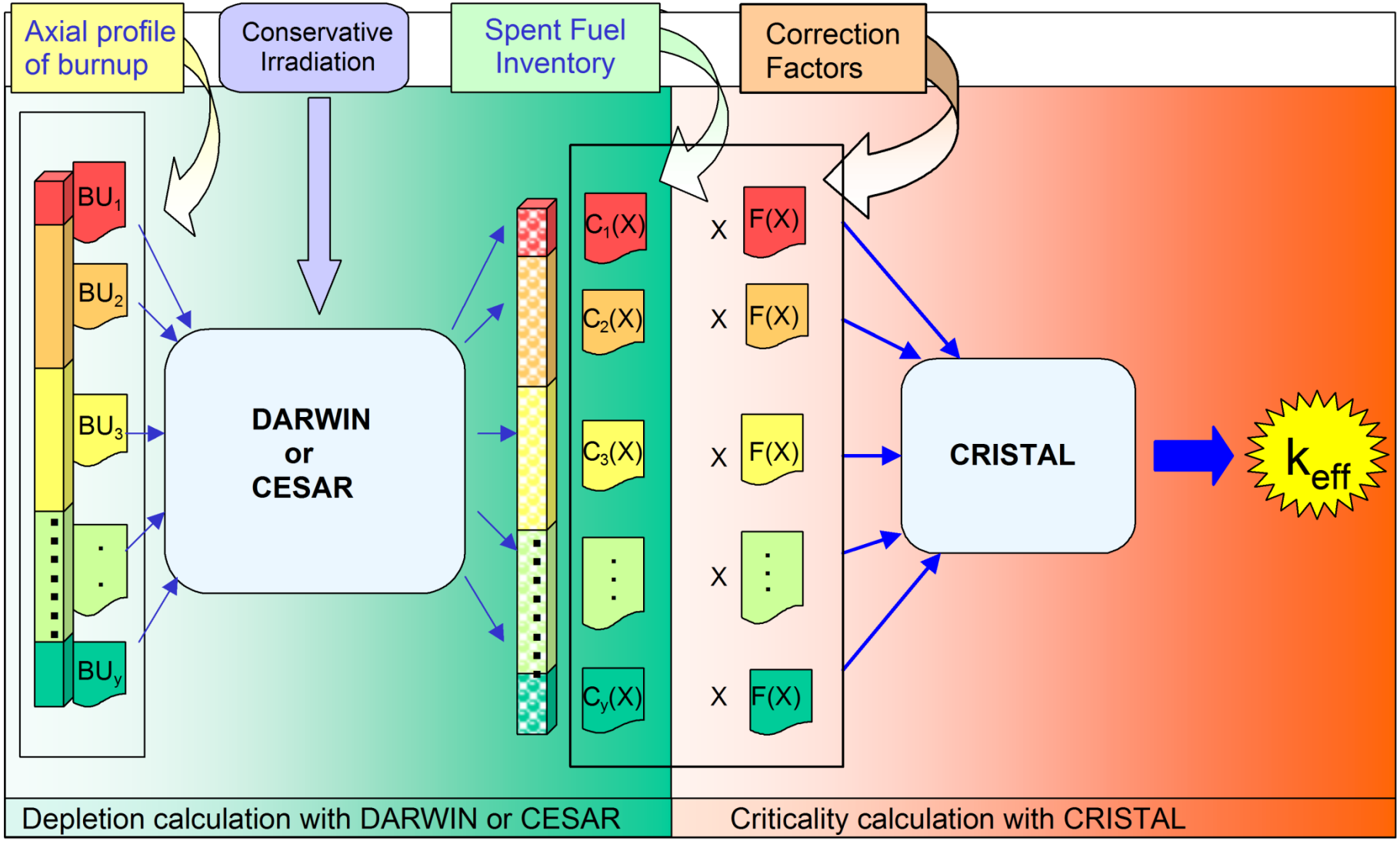
APPENDIX 3

EXPERIMENTAL VALIDATION DATABASE FOR DEPLETION CODES (2005)

EXPERIMENT Reactor	Fuel	Enrichment	Burnup	Experimental data
Tihange 1 1 to 3 startup cycles	PWR 15x15 UOX	3.10 wt%	10 to 40 GWd/MTHM	HN, MA, Nd Rods in center
Bugey 3 1 to 3 startup cycles	PWR 17x17 UOX	2.10 wt% 3.10 wt%	19 to 38 GWd/MTHM	HN, MA, Nd, Cs, BUC, Osc UF Rods in center
Fessenheim 2 2 startup cycles	PWR 17x17 UOX	2.60 wt%	27 to 30 GWd/MTHM	HN, MA, Nd Radial cross member of rods
Fessenheim 2 4 & 5 cycles	PWR 17x17 UOX	3.10 wt%	45 to 60 GWd/MTHM	HN, MA, Nd, Cs, Osc UF, µsensor
Gravelines 3+2 2 to 5 cycles	PWR 17x17 UOX	4.50 wt%	25 to 62 GWd/MTHM	HN, MA, Nd, Cs, BUC, SIMS, µsensor, Osc UF
Cruas 4 / ERU 1 to 3 cycles	PWR 17x17 Enriched UOX from 100% recycled uranium	3.61 wt% U6/U8 = 1.2 wt%	11 to 34 GWd/MTHM 4th cycle abandoned	HN, MA, Nd
Cruas 2 / High burnup 5 & 6 cycles	PWR 17x17 UOX extracted at 5 cycles, re-irradiated for 6th cycle	4.50 wt%	50 to 70 GWd/MTHM	HN, MA, Nd, Cs, BUC, SIMS, µsensor, Osc UF
Gravelines 5 / High burnup 5, 6 and 7 cycles	PWR 17x17 UOX re-irradiated for 6th and 7th cycle	4.50 wt%	55 to 80 GWd/MTHM	HN, MA, Nd, Cs, SIMS, µsensor, Osc UF
Saint Laurent B1 1 to 3 cycles 3 zones	PWR 17x17 MOX 3 Pu concentrations	Uapp. 0.22 wt% $\left(\frac{Pu}{U+Pu}\right)_{mean} \cong 4.5\ wt\%$	Sample of 3 zones and 3 BUs 10 to 45 GWd/MTHM	HN, MA, Nd, Cs, BUC, SIMS, µsensor, Osc UF
Gravelines 4 3 & 4 cycles Central zone and median zone	PWR 17x17 MOX 4th cycle against reflector	Uapp. 0.22 wt% $\left(\frac{Pu}{U+Pu}\right)_{mean} \cong 4.5\ wt\%$	40 to 50 GWd/MTHM	HN, Nd
Dampierre 2 / High burnup 1 to 5 cycles Central zone and median zone	PWR 17x17 MOX	Uapp. 0.22 wt% $\left(\frac{Pu}{U+Pu}\right)_{mean}$: 5.2 wt% and 6.6 wt%	Central zone: 1 to 5 cycles Median zone: 4, 5 cycles 10 to 53 GWd/MTHM	HN, MA, Nd, Cs, BUC, SIMS, µsensor, Osc UF
Sherwood 2 Mélusine	PWR simulation UO_2 doped with U^{236}	U5/U8 = 2.80 wt% 0.1 wt% < U6/U8 < 1.2 wt%	Central UO_2 ~ 2500 MWd/MTHM	Nuclides filiation, HN, MA, Nd
Gédéon 1 Mélusine	PWR simulation UO_2 + cr. appropriate Gd	UO_2 at 3.25 wt% Gd at 5 wt%	UO_2 from 3.0 to 10 GWd/MTHM	HN, Nd, Gd
Gédéon 2 Mélusine	PWR simulation UO_2 + Gd interaction	UO_2 at 3.25 wt% Gd at 8 wt%	UO_2 from 3.0 to 13 GWd/MTHM	HN, Nd, Gd, Osc I=UF
BWR rods Gundremmingen	BWR 9x9	UO_2 at 3.95 wt%	40 GWd/MTHM	HN, MA, Nd, BUC, µsensor, Osc UF

HN = heavy nuclei (U, Pu); **MA** = minor actinides (Np, Am, Cm); **BUC** = burnup credit (FP Prog.)
Osc UF = oscillation of used fuel samples in MINERVE; **µsensor** = electronic sensor, distribution of chemical elements; **SIMS** = nuclide distribution

APPENDIX 4: EXAMPLE OF CALCULATION SEQUENCING (USING THE CRISTAL V1 "STANDARD" ROUTE)



APPENDIX 5

USED FUEL INVENTORY EXPERIMENTAL VALIDATION

Table 9 to Table 15 present the results from the experimental validation of the used fuel inventory for PWR UOX fuels, calculated using the DARWIN V2.0 fuel cycle package, based on the CEA93-V6 library, and the APOLLO2.5 and PEPIN2.0 codes.

1. Uranium isotopes

Table 9: Summary of (C-E)/E biases (%) for the 'uranium' inventory

PWR fuel	BU (GWd/MTHM)	$^{234}U/^{238}U$	$^{235}U/^{238}U$	$^{236}U/^{238}U$
UOX BUGEY FESSENHEIM 3.1 wt%	20	2.5 ± 0.6	0.2 ± 1.3	-3.5 ± 1.0
	25	5.2 ± 1.1	0.06 ± 2.4	-3.1 ± 1.2
	40	1.7 ± 0.7	-0.1 ± 2.1	-3.2 ± 0.4
	50	-5.8 ± 1.1	0.2 ± 2.7	-4.1 ± 0.1
	60	-2.1 ± 2.3	4.2 ± 7.5	-3.9 ± 0.2
UOX GRAVELINES 4.5 wt%	30	-0.3 ± 0.9	-0.8 ± 1.2	-3.3 ± 1.0
	40	0.2 ± 1.3	-0.4 ± 2.8	-3.8 ± 1.1
	50	-4.6 ± 1.7	1.5 ± 4.1	-4.8 ± 0.7
	60	1.4 ± 1.0	1.1 ± 3.1	-4.3 ± 0.3
ERU CRUAS 3.5 wt%	15	0.3 ± 0.5	-0.2 ± 1.0	-0.5 ± 0.4
	25	0.2 ± 0.8	0.8 ± 1.7	-0.3 ± 0.3
	35	-0.4 ± 1.1	0.06 ± 3.1	-1.1 ± 0.2

2. Plutonium isotopes

Table 10: Summary of (C-E)/E biases (%) for the 'plutonium' inventory

PWR fuel	BU (GWd/MTHM)	$^{238}Pu/^{238}U$	$^{239}Pu/^{238}U$	$^{240}Pu/^{238}U$	$^{241}Pu/^{238}U$	$^{242}Pu/^{238}U$
UOX BUGEY FESSENHEIM 3.1 wt%	20	-12.6 ± 3.2	-0.6 ± 1.1	-0.7 ± 1.6	-3.6 ± 2.1	-6.0 ± 3.9
	25	-9.5 ± 4.6	-0.4 ± 1.5	-0.7 ± 2.1	-3.2 ± 2.9	-5.0 ± 5.0
	40	-6.9 ± 2.0	2.3 ± 0.9	-1.8 ± 0.8	-1.2 ± 1.2	-7.5 ± 2.2
	50	-5.0 ± 1.6	1.9 ± 1.1	-1.1 ± 0.6	-0.7 ± 1.2	-6.2 ± 1.8
	60	-9.2 ± 2.5	3.4 ± 2.4	-0.5 ± 0.9	-0.1 ± 2.6	-8.8 ± 2.9
UOX GRAVELINES 4.5 wt%	30	-13.1 ± 3.8	-1.5 ± 1.1	-2.0 ± 1.7	-4.9 ± 2.3	-6.7 ± 3.9
	40	-10.8 ± 5.1	-0.4 ± 1.8	-1.9 ± 2.0	-4.2 ± 2.7	-6.9 ± 4.8
	50	-11.1 ± 4.5	+1.6 ± 2.1	-1.7 ± 1.5	-3.1 ± 2.4	-8.6 ± 4.3
	60	-10.5 ± 2.5	+2.0 ± 1.4	-1.5 ± 0.7	-2.6 ± 2.2	-7.4 ± 2.2
ERU CRUAS 3.5 wt%	15	-3.4 ± 3.9	1.7 ± 1.2	-2.1 ± 2.6	0.3 ± 3.6	-5.3 ± 5.8
	25	-3.4 ± 3.4	1.0 ± 1.0	-0.5 ± 2.2	-1.8 ± 2.7	-5.0 ± 5.1
	35	-2.3 ± 3.3	1.9 ± 1.0	-0.3 ± 1.7	-0.7 ± 1.8	-4.3 ± 4.3

3. Neptunium 237, americium 241 and 243

Table 11: Summary of (C-E)/E biases (%) for the 'neptunium' and 'americium' inventories

PWR fuel	BU (GWd/MTHM)	$^{237}Np/^{238}U$	$^{241}Am/^{238}U$	$^{243}Am/^{238}U$
UOX BUGEY FESSENHEIM 3.1 wt%	20	-10.9 ± 4.4	-5.8 ± 3.4	-9.8 ±7.8
	25	-7.5 ± 4.3	-2.9 ± 3.1	-10.3± 7.5
	40	-1.5± 2.1	-3.2 ± 1.2	-4.6 ± 3.4
	50	-4.1 ± 2.5	-13.6 ± 2.6	-10.8 ± 2.9
	60	2.5 ± 3.1	0.02 ± 2.4	-5.5± 4.6
UOX GRAVELINES 4.5 wt%	30	-1.2 ± 4.7	-5.0 ± 2.8	-10.1 ± 7.7
	40	-3.2 ± 4.2	-3.5 ± 2.1	-8.0 ± 7.0
	50	-4.5 ± 3.4	/	/
	60	-3.7 ± 3.1	+1.3 ± 0.6	-4.8 ± 3.9
ERU CRUAS	15	-4.6 ± 3.2	/	/
	25	-0.7 ± 3.1	/	/
	35	3.3 ± 3.0	-1.4 ± 2.3	-5.1 ± 6.4

4. Neodymium 143 and cesium 133

Table 12: Summary of (C-E)/E biases (%) for the ^{143}Nd and ^{133}Cs inventories

PWR fuel	BU (GWd/MTHM)	$^{143}Nd/^{238}U$	$^{133}Cs/^{238}U$
UOX BUGEY FESSENHEIM 3.1 wt%	20	0.3 ± 1.3	-2.7 ± 2.0
	25	0.3 ± 1.7	-3.9 ± 1.9
	40	/	-8.5 ± 1.8
	50	2.4 ± 1.8	-1.3 ± 1.6
	60	/	-2.2 ± 1.5
UOX GRAVELINES 4.5 wt%	30	/	-4.9 ± 2.0
	40	0.5 ± 1.4	-3.3 ± 2.0
	50	1.6 ± 1.0	-3.9 ± 1.8
	60	1.9 ± 0.4	-3.4 ± 1.2
ERU CRUAS 3.5 wt%	15	0.0 ± 1.8	/
	25	0.6 ± 1.6	/
	35	0.7 ± 1.2	/

5. Metallic fission products: technicium 99, molybdenum 95, ruthenium 101, rhodium 103 and silver 109

Table 13: Summary of (C-E)/E biases (%) for the 'metallic FP' inventory

PWR fuel	BU (GWd/MTHM)	$^{99}Tc/^{238}U$	$^{95}Mo/^{238}U$	$^{101}Ru/^{238}U$	$^{103}Rh/^{238}U$	$^{109}Ag/^{238}U$
UOX BUGEY 3.1 wt%	20	-7.6 ± 3.1	-3.7 ± 2.7	2.9 ± 2.9	3.5 ± 2.8	-47.3 ± 4.0
	40	4.8 ± 3.0	8.4 ± 2.6	15.2 ± 2.9	13.9 ± 2.6	-13.7 ± 3.7
UOX GRAVELINES 4.5 wt%	40	-0.6 ± 3.1	5.5 ± 2.7	5.0 ± 2.9	3.4 ± 2.7	/
	50	2.6 ± 3.1	2.6 ± 2.6	6.4 ± 2.3	4.6 ± 2.5	/
	60	3.6 ± 3.0	2.1 ± 2.6	0.9 ± 2.9	3.0 ± 2.3	/

6. Gadolinium 155 and europium 153

Table 14: Summary of (C-E)/E biases (%) for the ^{155}Gd and ^{153}Eu inventories

PWR fuel	BU (GWd/MTHM)	$^{155}Gd/^{238}U$	$^{153}Eu/^{238}U$
UOX BUGEY 3.1 wt%	20	-3.1 ± 3.6	6.5 ± 3.0
	40	0.1 ± 4.1	11.8 ± 2.6
UOX GRAVELINES 4.5 wt%	40	4.3 ± 4.1	8.9 ± 3.0
	50	8.1 ± 4.3	11.7 ± 2.8
	60	11.9 ± 4.4	16.4 ± 2.5

7. Samarium isotopes

Table 15: (C-E)/E biases (%) for the 'samarium' inventory

PWR fuel	BU (GWd/MTHM)	$^{147}Sm/^{238}U$	$^{149}Sm/^{238}U$	$^{150}Sm/^{238}U$	$^{151}Sm/^{238}U$	$^{152}Sm/^{238}U$
UOX BUGEY 3.1 wt%	20	-4.9 ± 1.5	4.5 ± 4.7	-3.7 ± 2.6	-3.1 ± 1.4	0.4 ± 1.9
	40	-5.9 ± 1.0	-3.4 ± 23.3	-3.6 ± 2.2	7.3 ± 1.7	1.1 ± 1.7
UOX GRAVELINES 4.5 wt%	40	-6.5 ± 1.1	-8.6 ± 8.5	-3.2 ± 2.3	3.6 ± 1.5	1.4 ± 1.8
	50	-7.0 ± 1.0	5.6 ± 4.9	-4.8 ± 2.1	7.9 ± 1.7	3.5 ± 1.6
	60	-7.4 ± 0.9	-3.5 ± 10.1	-4.1 ± 1.8	12.8 ± 3.0	5.6± 1.7

APPENDIX 6

REACTIVITY WORTH EXPERIMENTAL VALIDATION

Table 16 presents the results from the experimental validation of reactivity worth for PWR UOX fuels, carried out with the CRISTAL V1 criticality safety package, based on the APOLLO2.5 code and its library, CEA93.V6.

The interpretation of the oscillation experiments on separate samples of fission products revised the overestimation of ^{99}Tc and ^{133}Cs (4% in R1UO2). The calculation over experiment biases are satisfactory with regard to the ^{153}Eu, ^{109}Ag, ^{152}Sm, ^{145}Nd and ^{155}Gd (underestimation of less than 2%). The ^{143}Nd and ^{95}Mo are underestimated by 4% in R1UO2, and the ^{149}Sm by 6%. With regard to the ^{103}Rh, the interpretation of the oscillation experiments showed a significant overestimation.

No information or trend could be ascertained for the reactivity worth of ^{147}Sm, ^{150}Sm, ^{151}Sm or ^{101}Ru, as the capture of these samples can be largely attributable to the residual ^{149}Sm or the ^{99}Ru with regard to the ^{101}Ru.

Table 16: Summary of (C-E)/E biases (%) for 11 burnup credit nuclides

Fission product	R1UO2	R2UO2
^{143}Nd	-4.5 ± 2.5	-10.0 ± 3.0
^{145}Nd	0.4 ± 4.1	-1.9 ± 4.8
^{149}Sm	-5.7 ± 2.1	-9.8 ± 2.5
^{152}Sm	-0.2 ± 3.2	-1.2 ± 4.2
^{95}Mo	-3.6 ± 3.4	-7.1 ± 3.8
^{103}Rh	+13.3 ± 4.0	+11.9 ± 3.8
^{155}Gd	-1.9 ± 2.9	-11.0 ± 3.6
^{153}Eu	-1.6 ± 4.4	-3.6 ± 5.1
^{109}Ag	-1.7 ± 4.0	-1.5 ± 3.6
^{99}Tc	4.2 ± 3.8	-2.2 ± 3.5
^{133}Cs	4.3 ± 1.9	2.0 ± 2.0

APPENDIX 7

DETERMINATION OF CORRECTION FACTORS

I PREAMBLE

At the Working Group meeting on 19 June 2003, CEA was assigned the task of establishing an approach to determine the correction factors to be applied to the nuclide concentrations to take account of the experimental validation of the depletion codes and the experimental validation of the actinides and fission products cross-sections.

This approach was presented at the Working Group meeting on 5 February **2004** and gave rise to a number of discussions. The entire approach is currently being analyzed by the Working Group and is therefore expected to evolve.

It is to note that the described approach does not reflect ongoing work.

II APPROACH PROPOSED BY CEA TO DETERMINE THE CORRECTION FACTORS

1) Approach

Determining the correction factors relies on the experimental validation of the used fuel composition (see Appendix 5) and the experimental validation of the reactivity worth (see Appendix 6), based on the following approach:

The calculation C - experiment E comparison for the total reactivity worth of a nuclide, from an depletion calculation followed by a criticality calculation, represented as (N x ρ), gives the relation $(N \times \rho)_{exp} = \left(\frac{E}{C}\right) \times (N \times \rho)_{calc}$.

Working from the assumption that the calculation - experiment comparison for the used fuel inventory and the reactivity worth are two independent parameters (the C/E values obtained are derived from two strictly different calculations), the ratio $\left(\frac{E}{C}\right)$ is the product of the experiment - calculation comparison for the used fuel inventory $\left(\frac{E}{C}\right)_{BM}$ and the reactivity worth $\left(\frac{E}{C}\right)_{\rho}$: $\left(\frac{E}{C}\right) = \left(\frac{E}{C}\right)_{BM} \times \left(\frac{E}{C}\right)_{\rho}$.

With regard to the absorbent nuclides, for the criticality studies it is conservative to have an underestimated calculated total reactivity worth, illustrated as follows:

$$(N \times \rho)_{exp} \geq (N \times \rho)_{calc}, \text{ or } \left(\frac{E}{C}\right) \geq 1, \text{ or } \left(\frac{E}{C}\right)_{BM} \times \left(\frac{E}{C}\right)_{\rho} \geq 1.$$

Thus, if $\left(\frac{E}{C}\right)_{BM}$ x $\left(\frac{E}{C}\right)_{\rho}$ < 1, the inventory of the absorbent nuclide is modified by applying a correction factor equal to $\left(\frac{E}{C}\right)_{BM}$ x $\left(\frac{E}{C}\right)_{\rho}$. If $\left(\frac{E}{C}\right)_{BM}$ x $\left(\frac{E}{C}\right)_{\rho}$ ≥ 1 then the inventory is not modified, and the correction factor is therefore equal to 1.

With regard to the fissile nuclides, for the criticality studies it is conservative to have an overestimated calculated total reactivity worth, illustrated as follows:

$$(N \times \rho)_{exp} \leq (N \times \rho)_{calc}, \text{ or } \left(\frac{E}{C}\right) \leq 1, \text{ or } \left(\frac{E}{C}\right)_{BM} \times \left(\frac{E}{C}\right)_{\rho} \leq 1.$$

Thus, if $\left(\frac{E}{C}\right)_{BM}$ x $\left(\frac{E}{C}\right)_{\rho}$ > 1, the inventory of the fissile nuclides is modified by applying a correction factor equal to $\left(\frac{E}{C}\right)_{BM}$ x $\left(\frac{E}{C}\right)_{\rho}$. If $\left(\frac{E}{C}\right)_{BM}$ x $\left(\frac{E}{C}\right)_{\rho}$ ≤ 1 then the inventory is not modified, and the correction factor is therefore equal to 1.

2) Determination of correction factors

a) Calculation over experiment bias for the used fuel inventory

For each nuclide, the calculation over experiment bias for the used fuel inventory $\left(\frac{E}{C}\right)_{BM}$ adopted to determine the correction factors corresponds to:

- either the mean value obtained across all of the results when no C/E-1 divergence is observed with the burnup, while ensuring a certain level of consistency between the results (for example, a negative C/E-1 is not taken into account in the mean if the set of results indicates a positive C/E-1);
- or the value that minimises the k_{eff} if there is a divergence with the burnup or if no trend is revealed by the set of results. In the case of a fissile nuclide, this corresponds to the lowest C/E value, and in the case of an absorbent nuclide, it corresponds to the highest;
- with regard to the ^{239}Pu, the C/E-1 biases for the used fuel composition tend to be slightly underestimated at low burnup, then overestimated from 30 GWd/MTHM. As this nuclide has a notable influence on the reactivity value, two correction factors are recommended: one for a burnup lower than 30 GWd/MTHM and the other for a burnup of between 30 GWd/MTHM and 60 GWd/MTHM.

b) Calculation over experiment bias for the reactivity worth

- For each FP, the experiment over calculation bias for the reactivity worth $\left(\frac{E}{C}\right)_{\rho}$ adopted to determine the correction factors, corresponds to the C/E values obtained from the $R1UO_2$ measurements, as these are higher than the C/E values obtained from the $R2UO_2$ measurements, and therefore involve a lower k_{eff}, which is not conservative.
  The reactivity worth of fission products ^{147}Sm, ^{150}Sm, ^{151}Sm and ^{101}Ru could not be validated with the oscillation experiments in the MINERVE reactor. For this reason, we used the capture cross-section and resonance integral uncertainty data provided by the JEF2.2 evaluation for these fission products.

- For each actinide, due to the absence of an experimental programme to validate the reactivity worth, the uncertainty data from the JEF2.2 evaluation are used to estimate $\left(\frac{E}{C}\right)_{\rho}$: for the fissile nuclides, the estimate was based on the uncertainty data for neutron multiplication ν and the fission and absorption cross sections; for the other nuclides, the estimate incorporated the uncertainty data for the capture cross-section and the resonance integral.

3) Presentation of correction factors

The calculation over experiment biases for the used fuel composition and for the reactivity worth used to calculate the correction factors, and the correction factors themselves, are presented in Table 17 for each burnup credit nuclide.

Table 17: (C/E)-1 biases (%) for the used fuel composition and reactivity worth, with associated correction factors

Nuclide	Reactivity worth (C/E)-1 (%)	Used fuel composition (C/E)-1 (%)	Correction factor
^{234}U	+3 %	+1 %	0.96
^{235}U	+1 %	-1 %	1
^{236}U	+4 %	-4 %	1
^{238}Pu	+2 %	-11 %	1
^{239}Pu	-1 %	-1 % if BU ≤ 30 GWd/MTHM +2 % if 60 ≥ BU > 30 GWd/MTHM	1.02 is BU ≤ 30 GWd/MTHM 1 if BU 60 ≥ BU > 30 GWd/MTHM
^{240}Pu	+2 %	-2 %	1
^{241}Pu	-1 %	-5 %	1.06
^{242}Pu	+3 %	-7 %	1
^{237}Np	+3 %	-4 %	1
^{241}Am	+3 %	+1 %	0.96
^{243}Am	+3 %	-10 %	1
^{143}Nd	-4 %	+2 %	1
^{145}Nd	+0.4 %	-0.5 %	1
^{147}Sm	+5 %	-7 %	1
^{149}Sm	-6 %	+6 %	1
^{150}Sm	+4 %	- 4 %	1
^{151}Sm	+3 %	+13 %	0.86
^{152}Sm	-0.2 %	+6 %	0.94
^{95}Mo	-4 %	+4 %	1
^{103}Rh	+13 %	+4 %	0.85
^{155}Gd	-2 %	+12 %	0.91
^{153}Eu	-2 %	+16 %	0.88
^{109}Ag	-2 %	-14 %	1
^{99}Tc	+4 %	+4 %	0.92
^{101}Ru	+30 %	+6 %	0.72
^{133}Cs	+4 %	- 4 %	1

III REMARKS

In 2007 it was agreed that further studies should be conducted, in order to:

- take measurement uncertainties into account;
- perform statistical processing on the standard deviation on the C/E scatter plot;
- take the changes in the C/E into account with the burnup;
- assess the validity of compensation between the validation of the used fuel composition and the validation of the reactivity worth[12].

[12] For example, for ^{149}Sm, the approach described above suggests a correction factor of F_c=1 while: $\left(1+\left(\frac{C-E}{E}\right)_{mat.balance}\right)$>1 and $\left(1+\left(\frac{C-E}{E}\right)_{criticality\,calc.}\right)$<1.

APPENDIX 8

DETERMINATION OF A BURNUP PROFILE FOR THE STUDIES BASED ON THE PROFILES OBSERVED AT LA HAGUE

I POSSIBLE APPROACHES

Throughout this appendix, $B_i(z)$ represents the lower limit and $B_s(z)$ represents the upper limit of the profile range.

Based on the observations made at La Hague, the Working Group chose three options for determining the profile to be used in the "generic" studies:

- the mean profile $P_m(z)$ of the profile range (P(z)) represented in Figure 11 (bold solid line), where $P_m(z) = \frac{\sum_N P(z)}{N}$;
- a "penalizing" profile within each profile range. However, the penalizing nature of a profile is difficult to maintain for all transport and storage conditions, etc. This profile can also be difficult to symmetrize;
- either the "bounding" profile, which is lower than the observed profiles, as shown in Figure 12 (bold dotted line), or $B_i(z)$ as defined above.

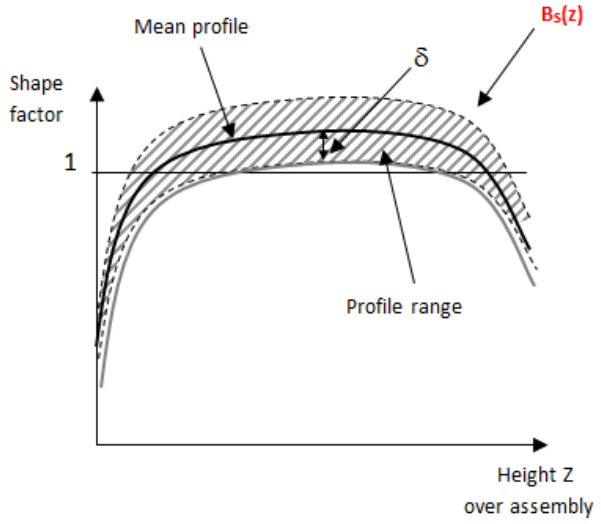


Figure 11: Mean burnup profile

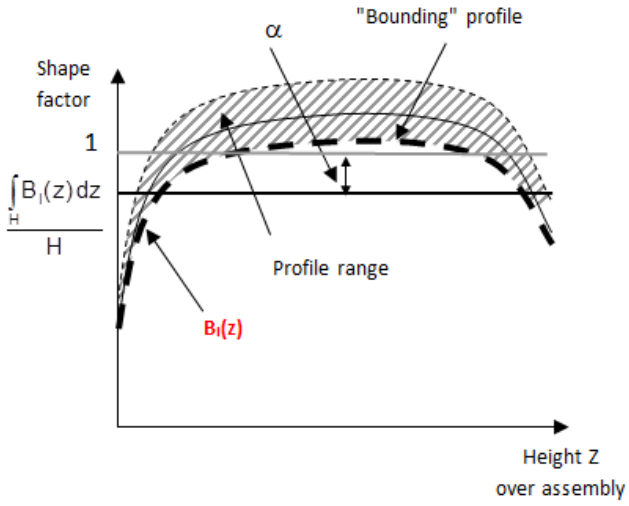


Figure 12: "Bounding" burnup profile

If the measurement methods could precisely determine the burnup then the following points should be noted:

- for a fixed mean burnup of an assembly BU_{actual}, a profile $P_{actual}(z)$ that is lower at one end than the mean profile ($P_m(z)$) of the profile range, could not be processed with a generic study using the "mean profile" and the burnup BU = BU_{actual} (or $BU_{actual} \times P_m(z)$), since at one end, the burnup $P_{actual}(z) \times BU_{actual}$ would be lower than that used in the study; all of the profiles observed within the range can be processed by a criticality study with the mean profile $P_m(z)$ but with a burnup reduced by the maximum burnup difference observed between the mean profile and the other profiles, i.e. BU-δBU (where δ is the maximum difference that exists between $P_m(z)$ and P(z); in fact δ = max ($B_s(z) - P_m(z)$); this is illustrated by a solid grey line in Figure 11;
- all of the burnup profiles observed in the range can be processed by a generic study using the "bounding profile" $B_i(z)$. However, for actual mean burnup profiles BU_{actual}, the mean burnup used in the study will then be lower than BU_{actual} (in fact, it will be equal to BU_{actual}-αBU_{actual}, where α, represented in Figure 12, has the value

$$1 - \frac{\int_H B_i(z)\, dz}{H})$$

II STUDY OF A MEAN PROFILE

1) Overview

For the studies conducted by the Burnup Credit Working Group, COGEMA gave EDF a database containing the measurements (taken in the T1 burnup pit at the plant at La Hague) for the axial burnup profile of 644 used fuel assemblies in EDF 1300 MWe reactors, grouped into two series, named R103 and R202, and reprocessed at La Hague. COGEMA provided the gross count rates for **^{137}Cs and ^{134}Cs** at centimetre intervals along two opposite sides (named route 1 and route 2 throughout the remainder of this document) of each of the assemblies and, deduced from these measurements, the mean burnups for each assembly and each side.

2) Description of the assemblies in the database

The majority of the assemblies reveals an initial ^{235}U enrichment of 3.10 wt%. The other assemblies, with an initial ^{235}U enrichment of 1.50 wt%, 1.80 wt%, 2.40 wt% and 2.95 wt%, are those from the first cores. These are associated with a fuel management with a balance enrichment of 3.10 wt%.

Over 94% of the assemblies have a burnup of between 30 and 40 GWd/MTHM. The other assemblies have a burnup of between 10 and 30 GWd/MTHM.

The table below shows the distribution of the assemblies according to their mean burnups, for each of the two series and each of the two routes.

Table 18: Population of assemblies based on their burnups for the series and routes studied

	Number of assemblies			
	Series R103 Route 1	**Series R103 Route 2**	**Series R202 Route 1**	**Series R202 Route 2**
BU < 25 GWd/MTHM	10	14	13	13
25 < BU < 30 GWd/MTHM	6	3	7	34
30 < BU < 40 GWd/MTHM	290	289	318	291
BU > 40 GWd/MTHM	0	0	0	0
Total assemblies	**306**	**306**	**338**	**338**

From the data provided by EDF/DPI/DPN/UNIPE, it can be seen that 33% of the measured assemblies were irradiated in the final cycle under control rods (R or G) and that 10% were irradiated under control rods for over two thirds of the time spent in the reactor. The database therefore contains a large number of assemblies placed under control rods during the irradiation process.

The representativeness of this database (assemblies with an initial ^{235}U enrichment equal to 3.1 wt% and a burnup of between 30 and 40 GWd/MTHM) remains to be confirmed, by providing axial burnup curve measurements for additional assemblies.

3) Formatting the data and establishing evaluation criteria

In order to compare the axial burnup curves, we formatted the initial data in advance. At each axial point (at centimetre intervals) of the active height of the assembly, we divided the local count rate by the total count rate and then multiplied the value by the height of the assembly to obtain a curve "centered" around the value 1. For each route, after removing the inconsistent points (negative or close to 0), and for each point, the measured local burnup was divided by the mean burnup value of the route studied, and by the number of measurement points.

The mean burnup profile, characteristic of each of the two series of assemblies and each of the routes, was established by calculating an arithmetic mean. The profile for the studied assembly was then compared to the calculated mean profile.

In order to quantify the level of consistency between an assembly profile and the mean profile for the series, two additional evaluation criteria were established:

- A Mean Standard Deviation (MSD) was associated with each burnup profile; this was defined for each assembly using the following formula:

$$\sigma_{plot} = \sqrt{\frac{\sum_i (\tau_i - \tilde{\tau}_i)^2}{N}}$$

where:

τ_i corresponds to the assembly burnup at point i;

$\tilde{\tau}_i$ corresponds to the mean profile burnup at point i;

N corresponds to the number of axial points along the total length of the assembly.

The summation is performed for all of the axial profile measuring points (centimetre by centimetre).

This criterion is used to evaluate the overall consistency of an assembly profile in relation to the mean profile. A profile is considered to be inconsistent with the mean profile when MSD > 10% (chosen as an arbitrary value).

A Series Mean Standard Deviation (SMSD) can be formulated based on this estimator, i.e. the mean of the mean standard deviations calculated for all of the assemblies for a series of burnups ranging between 30 and 40 GWd/MTHM.

- The Local Relative Deviation (LRD_i) is calculated at each axial point of an assembly and defined at each measuring point using the following formula:

$$LRD_i = \left| \frac{\tau_i - \tilde{\tau}_i}{\tilde{\tau}_i} \right|$$

The notations used are the same as those defined above.

This criterion is used to evaluate the local consistency, point by point, of an assembly profile in relation to the mean profile. A point is considered to be inconsistent with that of the mean profile when LRD_i > 20% (chosen as an arbitrary value). It should be noted that the level of uncertainty of the local burnup measurements, provided by COGEMA, was in the region of 20%.

For some assemblies, the burnup measured on one of the routes was higher than 30 GWd/MTHM, whereas on the other route it was lower than 30 GWd/MTHM. In the first case, the profile is taken into account when calculating the mean, while in the second case, it is rejected.

We were able to check that the inclusion of the profiles associated with mean burnups ranging between 25 and 30 GWd/MTHM, to produce the mean profile associated with a given series and route, would have very little effect on this. The MSD between the mean profile calculated for the assemblies with a burnup between 30 and 40 GWd/MTHM, and for those with a burnup between 25 and 40 GWd/MTHM, was always 0.2%, regardless of the series or the route.

4) Results from the analysis of series R103

Here we present the results for the assemblies that have a burnup of between 30 and 40 GWd/MTHM on both sides, representing 94.8% of series R103, and amounting to 290 assemblies.

- The mean standard deviation for route 1 of the series is ***4.9%***. We observe that, overall, none of the assemblies vary significantly from the mean profile for series R103-Route 1 (MSD > 10%). At the local level, three profiles for assemblies with an initial enrichment of 3.1 wt% and a burnup between 30 and 40 GWd/MTHM, vary significantly from the mean profile for series R103-Route 1 (with between 11 and 25 points displaying an LRD > 20%), representing 1% of the total number of assemblies with a burnup between 30 and 40 GWd/MTHM.

- The mean standard deviation for route 2 of the series is ***5.1%***. We observe that, overall, only one of the assemblies vary significantly from the mean profile for series R103-Route 2 (MSD > 10%). This anomaly is the result of an obvious measuring error. At the local level, with the exception of the profile affected by the measuring error, two assembly profiles vary significantly from the mean profile for series R103-Route 2 (with between 11 and 25 points displaying an LRD > 20%). It should be noted that the initial ^{235}U enrichment of one of these two assemblies is 3 wt%.

- The mean standard deviation between the normalized mean curves obtained for routes 1 and 2 is close to 1%. Furthermore, for series R103, we observe a mean relative deviation, or MRD (defined as the arithmetic mean for a series), of 3.6% from the relative deviation between the mean burnups for routes 1 and 2 on the same assembly.

A low mean standard deviation between the two mean curves, combined with a low MRD, demonstrates the close correlation between routes 1 and 2.

This close correlation between the results obtained for routes 1 and 2 on each assembly illustrates that the burnup across assemblies is similar.

5) Results from the analysis of series R202

Here we present the results specifically relating to the assemblies with a burnup of between 30 and 40 GWd/MTHM.

- The mean standard deviation for route 1 (318 assemblies) is 5.6%. We observe that, overall, none of the assemblies vary significantly from the mean profile for series R202-Route 1 (MSD > 10%).
  Similarly, at the local level, 19 profiles for assemblies with an initial enrichment of 3.1 wt% and a burnup between 30 and 40 GWd/MTHM, vary significantly from the mean profile for series R202-Route 1 (with between 11 and 25 points displaying an LRD > 20%), representing 6% of the total number of assemblies with a burnup between 30 and 40 GWd/MTHM.

- The mean standard deviation for route 2 (291 assemblies) is 5.8%. We observe that, overall, two of the assemblies vary significantly from the mean profile for series R202-Route 2 (MSD > 10%).
  Similarly, at the local level, with the exception of the profile affected by a measuring error, eight assembly profiles vary significantly from the mean profile for series R202-Route 2 (with between 11 and 25 points displaying an LRD > 20%). Among these eight assemblies, two also display a MSD > 10%.

The mean standard deviation between the normalized mean curves obtained for routes 1 and 2 is close to 0.6%. Furthermore, for series R202, we observe a mean relative deviation (MRD) of 4.00% for the burnups for routes 1 and 2.

This close correlation between the results obtained for routes 1 and 2 on each assembly illustrates that the burnup is similar on at least two sides of the assemblies.

6) Determination of an axial burnup curve representative of the database

Taking the assumption that the COGEMA database is representative of the vast majority of assemblies with an initial enrichment of 3.1 wt% and a burnup between 30 and 40 GWd/MTHM from 1300 MWe reactors, a mean curve can be defined.

The representativeness of this database remains to be confirmed, by providing axial burnup curve measurements for additional assemblies

Based on this feedback, we suggest establishing an axial burnup profile representative of the assemblies contained in the COGEMA database with initial ^{235}U enrichment equal to 3.1 wt% and with a mean burnup of between 30 and 40 GWd/MTHM. To do this, at each axial point (totalling 427 for a 1300 MWe PWR assembly) an arithmetic mean is calculated based on all of the selected profiles, corresponding to assemblies with an initial ^{235}U enrichment equal to 3.1 wt% and a mean burnup of between 30 and 40 GWd/MTHM. The mean profile obtained for each assembly is then normalized again.

This profile can be considered as being a mean model of the profiles from the COGEMA database.

To illustrate the low variability of the studied profiles, Figure 14Figure 14 shows the normalized mean curve, together with the maximum and minimum overall curves, obtained from each axial point of all of the assemblies with a mean burnup of between 30 and 40 GWd/MTHM.

With regard to the minimum and maximum overall curves, the mean standard deviations in relation to the mean curve are 18.05% and 17.23% respectively (remembering that the measurement uncertainty for the local burnups is 20%). For the minimum curve, these results were obtained based on 118 measuring points, 64 of which are located over the last 50cm of either end of the assembly; the mean relative deviation is greater than 20% (bearing in mind that a profile comprises 427 points). For the maximum curve, we took 95 measuring points, 53 of which are located over the last 50cm of either end of the assembly; the mean relative deviation is greater than 20% (bearing in mind that a profile comprises 427 points).

The table below describes the mean profile obtained, discretized according to 12 zones.

Table 19: Description of the mean profile discretized according to 12 zones.

Zone	Length (cm)	Beginning of zone	End of zone	Normalized BU
1	6	0	6	0.457
2	2	6	8	0.543
3	3	8	11	0.653
4	5	11	16	0.747
5	20	16	36	0.919
6	17	36	53	0.977
7	293.8	53	346.8	1.0618
8	35	346.8	381.8	1.014
9	17	381.8	398.8	0.935
10	20	398.8	418.8	0.736
11	5	418.8	423.8	0.545
12	3	423.8	426.8	0.476

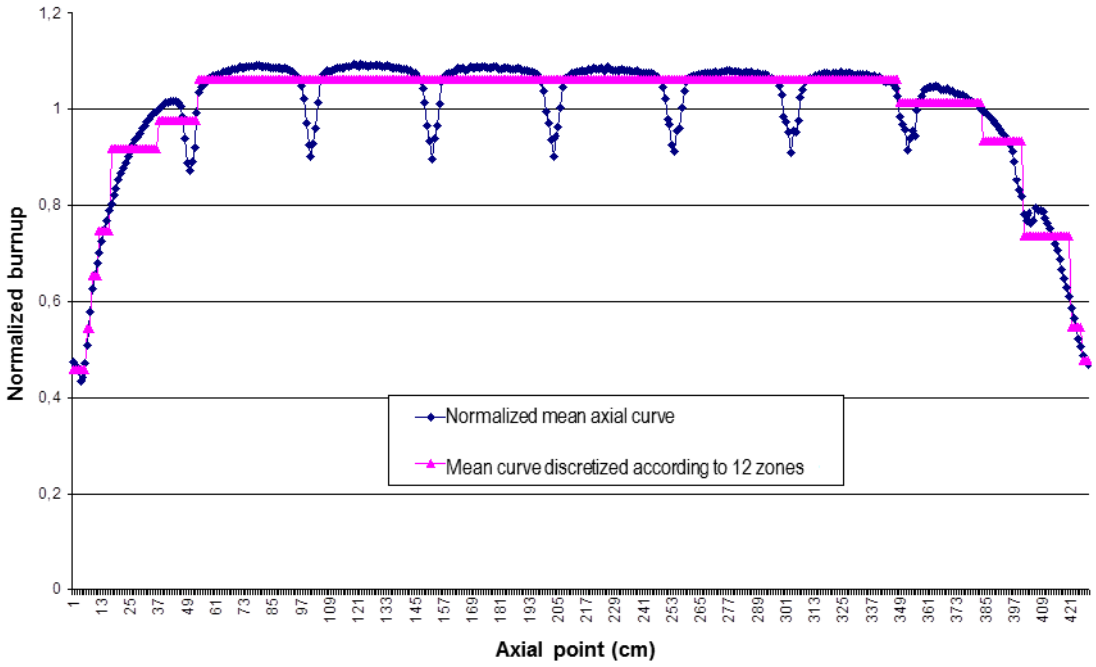


Figure 13: Normalized mean axial curves representative of an assembly with initial enrichment equal to 3.1 wt% and a burnup of between 30 and 40 GWd/MTHM from the COGEMA database, together with the mean curve discretized according to 12 zones

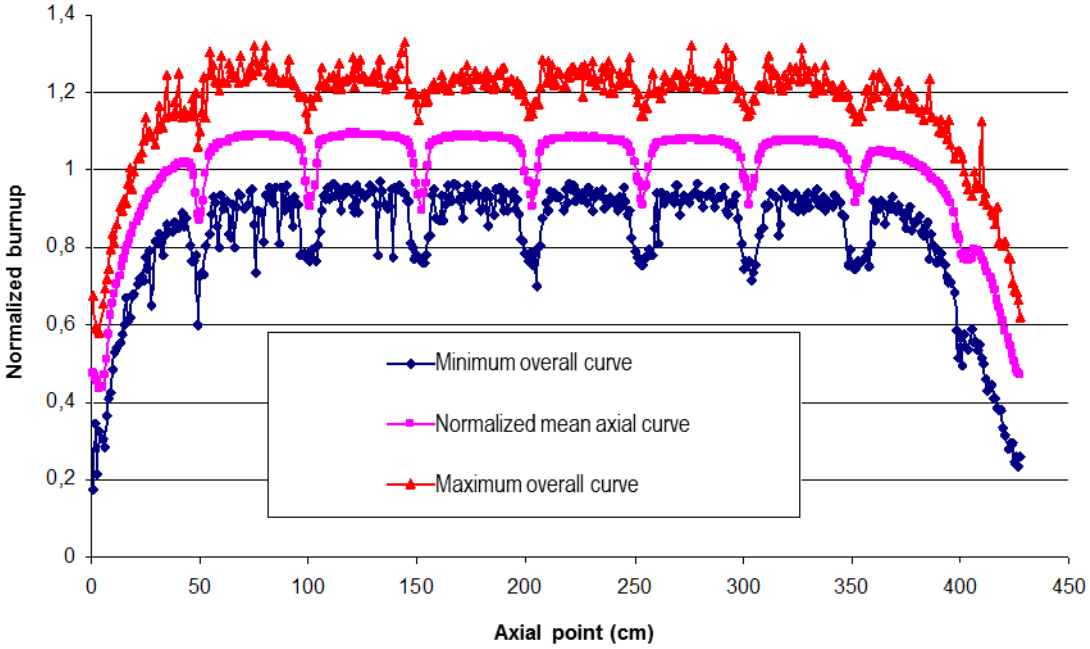


Figure 14: Normalized mean axial curves, and minimum and maximum overall curves, obtained from the assemblies with initial enrichment equal to 3.1 wt% and a burnup of between 30 and 40 GWd/MTHM from the COGEMA database

This profile is compared to the four mean burnup profiles characteristic of each of the two series and two routes (R103-Route 1, R103-Route 2, R202-Route 1 and R202-Route 2). The MSDs calculated in relation to the mean profile are 1.8%, 1.8%, 1.5% and 1.5% respectively. It can be seen that the MSDs for the mean profiles for series R202 are lower than those for series R103, since series R202 contained more assemblies than R103 (338 compared to 306), which significantly affect the mean profile.

In conclusion, the statistical analysis of the burnup profiles supplied by COGEMA does not reveal any particular axial deformation attributable to operation with partially inserted rods. The mean profile obtained is representative of the curves likely to be observed in French PWR operating modes used currently, or envisaged in the near future, including for the EPR. This representativeness will be extended to other axial burnup curve measurements for fuel assemblies. If there is no significant change in the operating mode for these reactors, and since the profile obtained is not found to be highly variable, there does not appear to be any reason why this profile should not be used in the criticality safety studies that take burnup credit into account. Using the mean profile without penalties for the criticality studies cannot be considered as a viable approach for taking burnup credit into account. Safety studies using such a profile will need to incorporate sufficient margins to ensure the overall conservatism of the approach.

APPENDIX 9

ISSUE OF COOLING TIME FOR LONG-TERM STORAGE

Variations in reactivity with the cooling time occur, on the one hand, as a result of changes in the characteristics of the used fuel composition (competition between the presence of fissile actinides, essentially ^{235}U, ^{239}Pu and ^{241}Pu, and absorbent actinides such as ^{241}Am, ^{240}Pu, ^{237}Np, ^{236}U and ^{238}U), and, on the other hand, due to variations in the neutron spectrum, which is closely linked to the present nuclides.

Although, on the whole, major trends can be easily ascertained (a significant decrease in reactivity during the first 100 years, an almost identical rise in amplitude over a period of between 20,000 and 40,000 years, followed by a further decrease over around 1 million years), the amplitudes of the negative and then positive variations are difficult to predict without prior calculations. They depend on both the irradiation conditions (i.e. the hypotheses relating to the calculation under flux) and the characteristics of the burnup profile used (i.e. the level of under-irradiation of the upper part).

For example, an assembly irradiated at between 10 and 20 GWd/MTHM with a flat profile, without the presence of rods, will be more reactive after 30,000 years of cooling than when it leaves the reactor. However, above 30 GWd/MTHM (with a flat profile), and all things being equal, the assembly will be most reactive when it leaves the reactor.

1. Short cooling times (< 1,000 years)

For handling or interim storage operations that take place during the first 1,000 years after the assembly has left the reactor, the issue is relatively easy to manage, in the sense that taking a fixed cooling time of 30 days into account will guarantee the conservative nature of the results.

However, margins may be determined if necessary, provided that the conservative nature of the selected cooling hypotheses is checked: if we take a cooling time T, the reactivity obtained at the time T must be greater than that recorded at the times t1 and t2, corresponding respectively to the beginning and end of the operations being studied, as shown in the figures below.

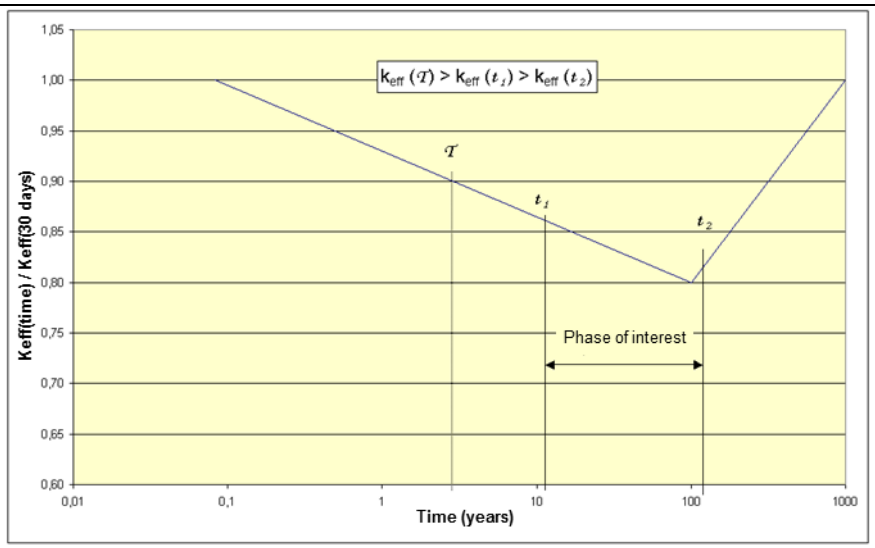


Figure 15: Example of use of the ratio $k_{eff}(t)/k_{eff}(30\ days)$

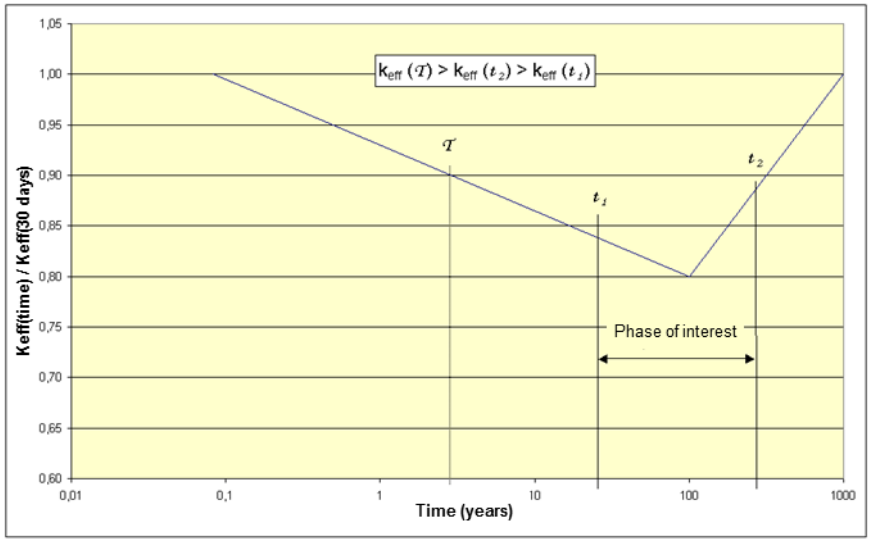


Figure 16: Example of use of the ratio $k_{eff}(t)/k_{eff}(30\ days)$

2. Long cooling times (> 1,000 years)

For handling or interim storage operations that take place more than 1,000 years after the assembly has left the reactor, the possibility of an increase in reactivity, above that seen when the reactor was shut down, demands that great care be taken when defining the cooling hypotheses, given that the reactivity peaks after 20,000 to 40,000 years. The use of a fixed cooling time of 30 days will not guarantee the conservative nature of the results.

It is clear that the characteristics of the used fuel composition after irradiation will condition the changes in reactivity with the cooling and will, therefore, influence whether or not the reactivity peaks at a higher level than initially.

Given the diversity of the input parameters (enrichment, irradiation, profiles) and the times in question, we maintain the principle that checking the conservative nature of the cooling hypotheses will be necessary, as illustrated by the figures below.

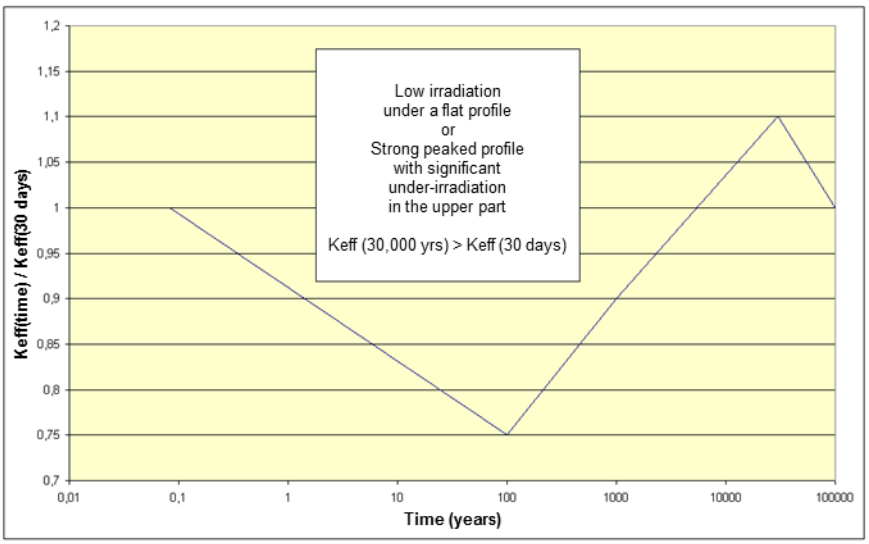


Figure 17: Example of checking the conservative nature of the cooling hypotheses

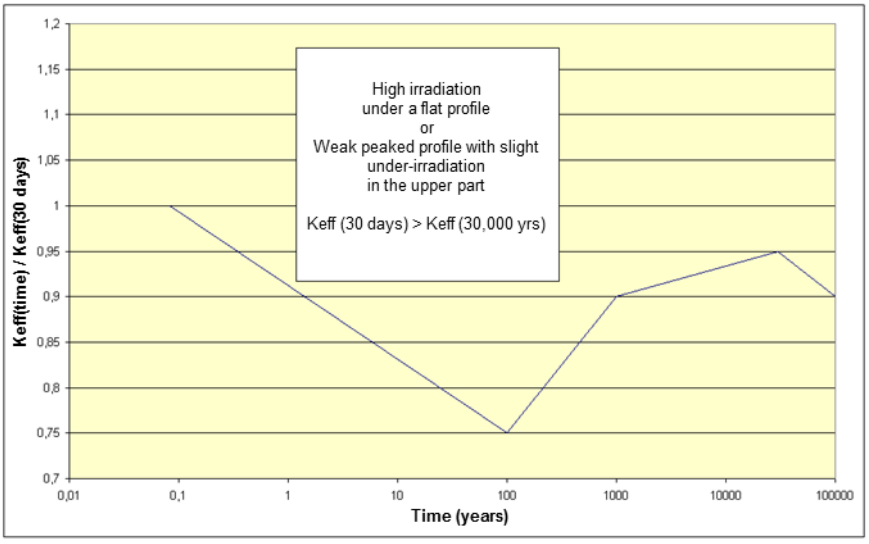


Figure 18: Example of checking the conservative nature of the cooling hypotheses

It can be ascertained that the transition to one curve or the other occurs at around 20 GWd/MTHM with a flat profile and 40 GWd/MTHM with a "Maillot" type peaked profile (below these values keff (30,000 years) > keff (30 days), and above these values keff (30 days) > keff (30,000 years)).

APPENDIX 10

IMPACT OF THE CONTROL ROD INSERTION HISTORY

I FEEDBACK

The diversity of the various plant series operated by EDF, and their associated fuel management systems, means that there is a whole range of fuel assemblies with relatively dispersed enrichments, especially when the startup cores have been loaded with assemblies other than those nominally enriched in accordance with the core management.

The table below summarizes some of this diversity.

Table 20: EDF core management

Plant series	CP0 (FE & BU)	CPY	1300	N4
Original management	1/3 3.25 wt%	1/3 3.25 wt%	1/3 3.10 wt%	1/3 3.40 wt%
Intermediate management	1/4 3.7 wt%	1/3 3.7 wt% 1/4 3.7 wt% 1/3 MOX	/	/
Current management	1/3 4.2 wt% CYCLADES	1/4 3.7 wt% MOX hybrid GARANCE	1/3 4.0 wt% GEMMES	Original management

The use of theoretical loading patterns to perform a statistical probability analysis of the presence of a control rod (CR)[13] during the life of an assembly is delicate, as the assemblies at the sites are not necessarily associated with a particular core management; switches can be made from one core management to another through so-called transition plans, which are different from the equilibrium plans; actual plans can differ from the theoretical plans depending on the actual availability of the assemblies; unplanned outages can lead the operator to rearrange the core, etc.

Consequently, rather than analyzing theoretical loading patterns, the statistical study presented here relates to the 38,821 PWR UOX assemblies supplied to La Hague in early 2005. Overall (with all assemblies combined), the following results are obtained:

- ✓ 51.5% of the population was irradiated with CRs for at least one reactor campaign;
- ✓ 22.8 % of the population was irradiated with CRs during the last reactor cycle.

[13] A control rod is taken to mean a fine control rod or a regulating rod; the term does not cover rods associated with reactor trip systems.

A slightly closer analysis could be conducted to reveal the trends associated with a particular type of assembly.

However, to preserve populations containing a significant number of objects, some of the assemblies are intentionally disregarded, such as those used on the startup cores (1.8 wt%, 2.1 wt%, 2.4 wt%, etc.), those that are too specific (including gadolinium), etc., in order to focus on the families that could be characterized by the nominal enrichments of the different core managements (3.25 wt%, 3.7 wt%, etc.). The results obtained are summarized in the table below.

Table 21: Proportion of assemblies with a control rod inserted during irradiation

Plant series	Enrichment	Population (assemblies)	% with CR inserted for at least one campaign	% with CR inserted in last cycle	% with CR inserted for all cycles
900 PWR	3.25 wt%	10,444	56.5 %	24.9 %	0.9 %
	3.70 wt%	9,375	62.9 %	12.4 %	0.4 %
1300 PWR	3.10 wt%	7,445	42.2 %	22.2 %	0.1 %
	4.00 wt%	1,263	52.3 %	0 %	0 %
N4	3.40 wt%	-	-	-	-

In relation to the overall statistics, although the major tendencies are effectively preserved (around 50% of the assemblies had control rods inserted, which took place during the last reactor cycle in around 50% of the cases), the following specific observations can be made:

- ✓ for fewer than 25% of the 3.7% assemblies irradiated with control rods, this occurred during their last reactor cycle;
- ✓ no assemblies under the GEMMES core management (1300 MWe PWR, E = 4 wt%) were irradiated with control rods during their last cycle in the reactor;
- ✓ only a very small number of assemblies were irradiated with control rods during all of the irradiation cycles (< 1%).

In conclusion, the feedback on the assemblies sent by EDF to La Hague in early 2005 reveals the following:

- ✓ a very large population was irradiated with control rods during at least one reactor campaign (around 52% of the assemblies). A close analysis of the results does not really reveal any particular category of assemblies that do not display this characteristic;

- ✓ the percentage of assemblies irradiated with control rods during the last reactor cycle is significant (around 23%). However, a close analysis of the results reveals some peculiarities: among the 900 MWe PWR assemblies enriched at 3.7 wt%, less than 13% of the population was irradiated with a control rod during the last reactor cycle, and this percentage was zero for the 1300 MWe PWR assemblies enriched at 4 wt%.

Moreover, when assemblies were irradiated with control rods, in general they were only inserted partially. The partial insertion of an absorbent rod is likely to have a significant impact on the results, due to both an under-irradiation effect on the upper part of the assembly, and a neutron spectrum hardening effect, which tends to favor the production of plutonium locally.

II IMPACT OF THE CONTROL ROD INSERTION HISTORY

The purpose of this study is to examine the impact on a criticality calculation of the irradiation of UOX assemblies with control rods (comprising 24 absorbent rods) inserted. To do this, depletion calculations were performed using the DARWIN V2 fuel cycle package, followed by a criticality calculation on a pool storage configuration using the CRISTAL V1 criticality safety package (the chosen criticality configuration maximizes the effects under examination).

To this end, three configurations are modelled in the depletion calculation:

1. central UOX assembly irradiated without control rods, surrounded by eight UOX assemblies irradiated without control rods (serving as a reference);
2. central UOX assembly irradiated with control rods inserted, surrounded by eight UOX assemblies irradiated without control rods (control rod insertion study);
3. central UOX assembly irradiated with control rods inserted, surrounded by eight MOX assemblies irradiated without control rods (MOX environment and control rod insertion study).

In this study, the used fuel composition for the criticality calculation consisted of the nuclides selected for the burnup credit studies, as follows:

- actinides ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{237}Np, ^{241}Am and ^{243}Am;
- the 15 fission products: ^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{152}Sm, ^{155}Gd, ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{109}Ag, ^{145}Nd, ^{147}Sm, ^{150}Sm, ^{151}Sm and ^{153}Eu;
- with a zero cooling time.

In order to maximize the effect of the control rods, we used the most efficient absorbent, B_4C (it should be noted that in the French 900 MWe PWRs, the rods consist of 24 silver-indium-cadmium rods).

In order to separate the used fuel composition problem from the "end effect" problem, the axial burnup profile was taken to be uniform and equal to the assembly mean burnup.

In the tables below, $\Delta\rho$ is the difference in reactivity defined by the formula:

$$\Delta\rho = \rho\,(\text{case2}) - \rho\,(\text{case1}) = Ln\frac{K_{eff}\,(\text{case2})}{K_{eff}\,(\text{case1})}$$

1) Impact of the FULL axial insertion of control rods

a) In the central UOX assembly surrounded by UOX

Table 22 shows the effect of the full axial insertion of control rods on the reactivity of a lattice of UOX assemblies, throughout the entire irradiation process or during a part of it, for each assembly in the lattice.

Table 22: Effect of irradiation on the reactivity of a UOX lattice, in a UOX environment with a full axial insertion of control rods compared to a UOX environment without control rods

	ρ (UOX with CR, UOX env.) - ρ (UOX w/out CR, UOX env.)			
BU of UOX assembly lattice	**UOX with CR from 0 to 40 GWd/MTHM**	**UOX with CR from 0 to 12 GWd/MTHM**	**UOX with CR from 12 to 24 GWd/MTHM**	**UOX with CR from 24 to 40 GWd/MTHM**
10 GWd/MTHM	+ 274 pcm	+ 274 pcm	-	-
20 GWd/MTHM	+ 1,067 pcm	+ 471 pcm	+ 419 pcm	-
30 GWd/MTHM	+ 2,260 pcm	+ 444 pcm	+ 814 pcm	+ 481 pcm
40 GWd/MTHM	+ 4,053 pcm	+ 516 pcm	+ 675 pcm	+ 2,085 pcm

The presence of control rods during the irradiation of a UOX assembly hardens the neutron spectrum, leading to an increase in the production of plutonium and, therefore, an increase in the reactivity of the UOX assembly lattice. This increase is all the more considerable when the control rods are inserted for a long period of time or at a later point during the irradiation process. The maximum increase in reactivity is around 4,000 pcm if the control rods are fully inserted throughout the entire irradiation (up to 40 GWd/MTHM).

b) In the central UOX assembly surrounded by MOX

Table 23 shows the effect of the full axial insertion of control rods on the reactivity of a lattice of UOX assemblies, throughout the entire irradiation of each assembly.

Table 23: Effect of irradiation on the reactivity of a UOX lattice, in a MOX environment with the full axial insertion of control rods compared to a MOX environment without control rods

BU of UOX assembly lattice	ρ (UOX with CR, MOX env) - ρ (UOX w/out CR, MOX env.)
10 GWd/MTHM	+ 364 pcm
20 GWd/MTHM	+ 1,195 pcm
30 GWd/MTHM	+ 2,421 pcm
40 GWd/MTHM	+ 3,991 pcm

If we compare these values to those in Table 1, it can be seen that the effect of the control rods insertion is the same.

Table 24 shows the effect of irradiation on the reactivity of a UOX lattice, in a MOX environment with the full axial insertion of the control rods compared to a UOX environment without control rods.

Table 24: Effect of irradiation on the reactivity of a UOX lattice, in a MOX environment with the full axial insertion of the control rods compared to a UOX environment without control rods

	ρ (UOX with CR, MOX env.) - ρ (UOX w/out CR, UOX env.)	
BU of UOX assembly lattice	UOX with CR from 0 to 40 GWd/MTHM	UOX with CR from 24 to 40 GWd/MTHM
10 GWd/MTHM	+ 408 pcm	-
20 GWd/MTHM	+ 1,494 pcm	-
30 GWd/MTHM	+ 3,093 pcm	-
40 GWd/MTHM	+ 5,242 pcm	+ 3,280 pcm

When the effects of irradiation with the control rods inserted (Table 1) and a MOX environment (Appendix 2) are combined, this leads to an increase in reactivity of up to

5,200 pcm for a lattice of UOX assemblies irradiated at 40 GWd/MTHM, assuming that all of the assemblies had the control rods fully inserted throughout the entire irradiation process in a complete MOX environment (eight MOX assemblies).

2) Impact of the partial axial insertion of the control rods (in the central UOX assembly surrounded by UOX assemblies)

In reality, control rods are only inserted partially during the operation of French reactors. In order to quantify the impact of the partial insertion of control rods, we model the insertion of control rods in the upper part of the fuel assembly, to an axial depth of 70 cm, for the criticality calculation (see Figure 19).

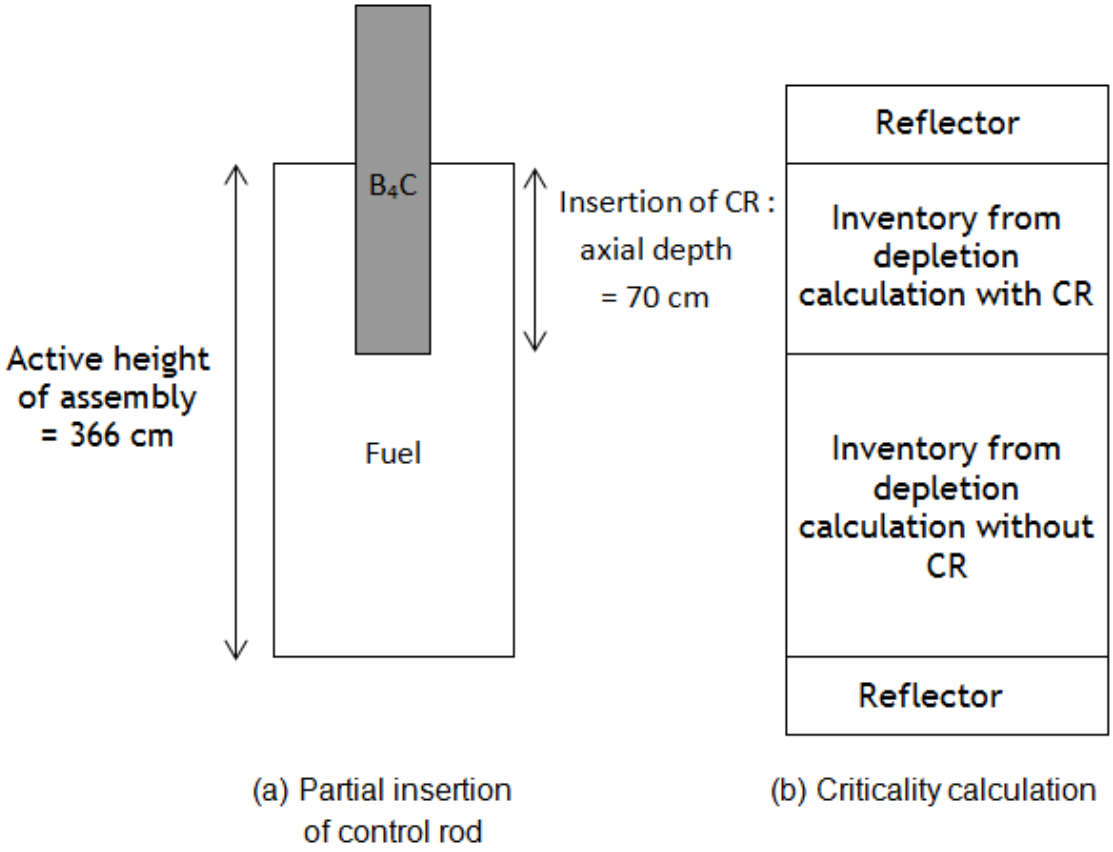


Figure 19: Illustration of the criticality calculation for the partial insertion of control rods

The impact on an infinite lattice of UOX assemblies, irradiated at 40 GWd/MTHM with a partial insertion (70cm) of the control rods, throughout the irradiation of each UOX assembly, is around +1,300 pcm (compared to +4,000 pcm with the full axial insertion of the control rods).

APPENDIX 11

STUDY OF POISONED RODS FOR 1300 MWE PWR ASSEMBLIES

1. PRESENTATION OF THE STUDY

In France, certain UOX management methods, such as CYCLADES for the 900 MWe PWRs or GEMMES for the 1300 MWe PWRs, include assemblies reloads containing gadolinium rods.

As part of the studies on burnup credit, the impact on the used fuel composition, and on the infinite multiplication factor (outside the reactor), of the presence of gadolinium rods during the irradiation of a UOX assembly is evaluated.

The case of a 17x17 PWR assembly with four or eight gadolinium rods is studied, with an initial enrichment of the UO_2 rods of 4.1 wt%. The ^{235}U enrichment of the Gd_2O_3 rods is 0.72 wt% with a Gd_2O_3 enrichment of 8 wt%, or 2.6 wt% with a Gd_2O_3 enrichment of 6 wt%.

The positions of the four and eight gadolinium rods in the UOX assembly are shown in Figure 20.

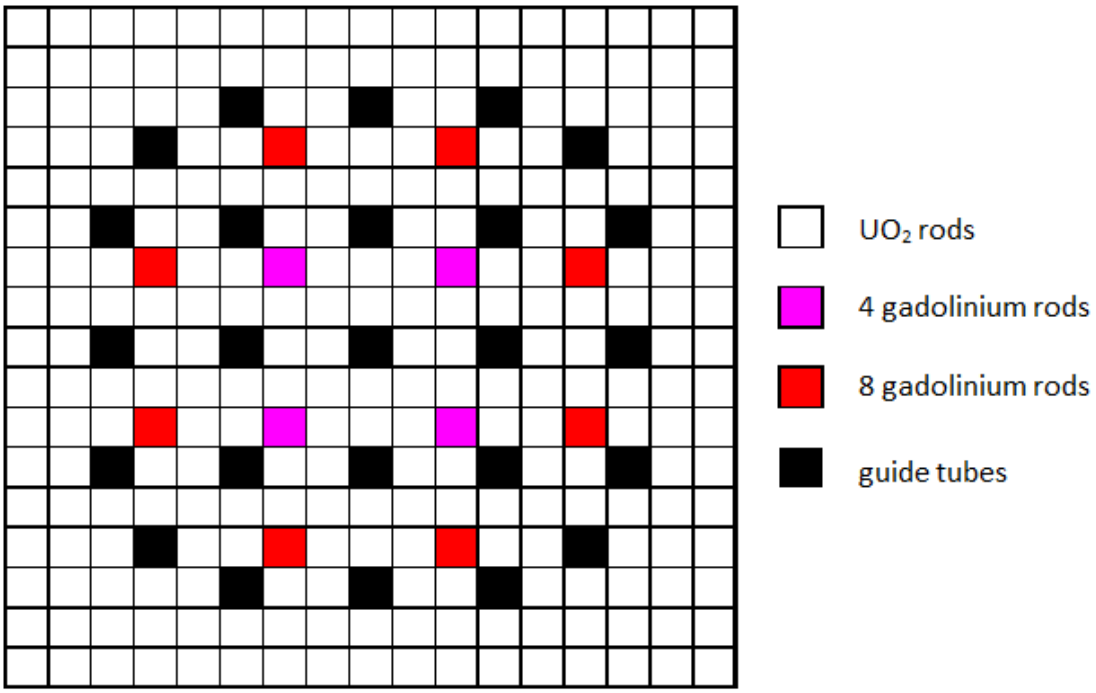


Figure 20: Positions of the four and eight gadolinium rods in the PWR UOX assembly

The depletion calculations are performed using the APOLLO2.5 code and the criticality calculations are performed with the CRISTAL V1.0 criticality safety package.

2. IMPACT ON THE USED FUEL INVENTORY

2.1 Four gadolinium rods with 0.72 wt% ^{235}U

Table 25 shows the difference in the used fuel inventory of the BUC nuclides between irradiation with four gadolinium rods (0.72 wt% ^{235}U) in the assembly and irradiation without gadolinium rods.

Table 25: Four Gd_2O_3 0.72 wt% - Observed biases (%) in the used fuel composition

BU (GWd/MTHM) No cooling	0	2	5	10	20	30	40	50	60
^{234}U	-1.5%	-1.6%	-1.7%	-1.8%	-1.9%	-2.0%	-2.1%	-2.2%	-2.2%
^{235}U	-1.28%	-1.32%	-1.39%	-1.51%	-1.75%	-2.06%	-2.47%	-2.86%	-3.24%
^{236}U		-0.1%	-0.2%	-0.3%	-0.6%	-0.7%	-0.8%	-1.0%	-1.1%
^{238}U	-0.107%	-0.108%	-0.112%	-0.118%	-0.122%	-0.126%	-0.131%	-0.135%	-0.139%
^{238}Pu		4.1%	4.1%	3.7%	2.0%	1.2%	0.7%	0.3%	0.0%
^{239}Pu		2.0%	1.9%	1.7%	0.4%	-0.1%	-0.4%	-0.6%	-0.5%
^{240}Pu		2.1%	1.8%	2.1%	1.7%	1.2%	0.8%	0.4%	0.1%
^{241}Pu		4.8%	3.5%	1.8%	1.6%	0.8%	0.3%	0.1%	-0.1%
^{242}Pu		5.4%	4.5%	3.6%	2.4%	2.1%	1.8%	1.5%	1.1%
^{237}Np		2.3%	2.2%	1.8%	0.6%	0.1%	-0.2%	-0.4%	-0.6%
^{241}Am		4.9%	4.1%	3.0%	1.5%	0.9%	0.2%	-0.2%	-0.5%
^{243}Am		8.1%	7.4%	6.2%	3.2%	2.6%	2.2%	1.8%	1.5%
^{103}Rh		-0.3%	-0.3%	-0.2%	0.1%	0.1%	0.1%	0.0%	-0.1%
^{133}Cs		-0.4%	-0.4%	-0.3%	-0.2%	-0.2%	-0.2%	-0.2%	-0.2%
^{143}Nd		-0.5%	-0.5%	-0.4%	-0.4%	-0.4%	-0.5%	-0.6%	-0.7%
^{145}Nd		-0.4%	-0.4%	-0.4%	-0.3%	-0.3%	-0.3%	-0.3%	-0.3%
^{155}Gd (FP)		0.2%	1.1%	1.2%	0.1%	-0.3%	-0.6%	-1.0%	-0.4%
^{95}Mo		-0.5%	-0.5%	-0.5%	-0.3%	-0.3%	-0.3%	-0.3%	-0.3%
^{99}Tc		-0.4%	-0.4%	-0.3%	-0.2%	-0.2%	-0.2%	-0.2%	-0.2%
^{101}Ru		-0.4%	-0.3%	-0.2%	-0.1%	0.0%	0.0%	0.0%	-0.1%
^{109}Ag		1.1%	1.4%	1.6%	1.6%	1.3%	1.0%	0.8%	0.6%
^{147}Sm		-0.5%	-0.6%	-0.7%	-0.6%	-0.6%	-0.6%	-0.7%	-0.7%
^{149}Sm		0.3%	1.1%	0.3%	-0.8%	-0.9%	-1.0%	-1.7%	-0.9%
^{150}Sm		-0.4%	-0.3%	0.0%	0.1%	0.1%	0.1%	0.0%	-0.1%
^{151}Sm		-0.4%	-0.2%	0.1%	-0.4%	-0.5%	-0.6%	-0.6%	-0.5%
^{152}Sm		-0.3%	-0.3%	-0.3%	0.1%	0.1%	0.1%	0.1%	0.0%
^{153}Eu		0.0%	0.4%	0.7%	0.5%	0.4%	0.4%	0.2%	0.1%

2.2 Eight gadolinium rods with 0.72 wt% ^{235}U

Table 26 shows the difference in the used fuel inventory of the BUC nuclides between irradiation with eight gadolinium rods (0.72 wt% ^{235}U) in the assembly and irradiation without gadolinium rods.

Table 26: Eight Gd_2O_3 0.72 wt% - Observed biases (%) in the used fuel composition

BU (GWd/MTHM) No cooling	0	2	5	10	20	30	40	50	60
^{234}U	-3.0%	-3.1%	-3.3%	-3.5%	-3.7%	-3.8%	-3.9%	-4.0%	-4.0%
^{235}U	-2.56%	-2.64%	-2.75%	-2.92%	-3.24%	-3.63%	-4.06%	-4.45%	-4.67%
^{236}U		-0.2%	-0.5%	-0.8%	-1.4%	-1.7%	-1.9%	-2.2%	-2.3%
^{238}U	-0.21%	-0.22%	-0.23%	-0.24%	-0.25%	-0.26%	-0.27%	-0.28%	-0.29%
^{238}Pu		8.7%	8.4%	7.1%	3.4%	1.7%	0.8%	0.2%	-0.3%
^{239}Pu		4.9%	4.6%	3.9%	1.6%	0.6%	0.2%	0.1%	0.3%
^{240}Pu		5.0%	4.3%	4.3%	3.6%	2.7%	2.0%	1.5%	1.0%
^{241}Pu		11.0%	8.7%	5.6%	3.4%	2.0%	1.2%	0.7%	0.6%
^{242}Pu		11.9%	10.0%	8.0%	5.3%	4.1%	3.3%	2.6%	1.9%
^{237}Np		4.8%	4.5%	3.5%	0.9%	-0.1%	-0.7%	-1.0%	-1.3%
^{241}Am		11.1%	9.5%	7.2%	4.0%	2.4%	1.3%	0.7%	0.5%
^{243}Am		17.6%	15.7%	12.6%	6.8%	5.1%	4.0%	3.1%	2.4%
^{103}Rh		-0.6%	-0.5%	-0.3%	0.2%	0.3%	0.3%	0.2%	0.1%
^{133}Cs		-0.9%	-0.9%	-0.8%	-0.5%	-0.5%	-0.4%	-0.5%	-0.5%
^{143}Nd		-1.0%	-1.0%	-1.0%	-0.9%	-0.9%	-1.0%	-1.0%	-1.0%
^{145}Nd		-0.9%	-1.0%	-0.9%	-0.8%	-0.7%	-0.7%	-0.7%	-0.7%
^{155}Gd (FP)		0.7%	2.9%	3.6%	0.6%	0.1%	-0.2%	-1.3%	0.2%
^{95}Mo		-1.0%	-1.0%	-1.0%	-0.8%	-0.8%	-0.7%	-0.7%	-0.7%
^{99}Tc		-0.9%	-0.9%	-0.8%	-0.6%	-0.5%	-0.4%	-0.4%	-0.5%
^{101}Ru		-0.8%	-0.7%	-0.5%	-0.3%	-0.2%	-0.2%	-0.2%	-0.2%
^{109}Ag		2.6%	3.4%	3.7%	3.4%	2.7%	2.1%	1.7%	1.3%
^{147}Sm		-1.0%	-1.2%	-1.4%	-1.3%	-1.1%	-1.1%	-1.1%	-1.0%
^{149}Sm		0.8%	2.3%	0.8%	-1.3%	-1.2%	-1.2%	-1.4%	-0.6%
^{150}Sm		-0.9%	-0.6%	-0.2%	0.0%	0.0%	-0.1%	-0.1%	-0.2%
^{151}Sm		-0.7%	-0.3%	0.3%	-0.6%	-0.7%	-0.7%	-0.7%	-0.4%
^{152}Sm		-0.6%	-0.7%	-0.6%	0.1%	0.1%	0.1%	0.0%	-0.1%
^{153}Eu		0.0%	0.7%	1.2%	0.8%	0.6%	0.5%	0.3%	0.2%

2.3 Four gadolinium rods with 2.6 wt% ^{235}U

Table 27 shows the difference in the used fuel inventory of the BUC nuclides between irradiation with four gadolinium rods (2.6 wt% ^{235}U) in the assembly and irradiation without gadolinium rods.

Table 27: Four Gd_2O_3 2.6 wt% - Observed biases (%) in the used fuel composition

BU (GWd/MTHM) No cooling	0	2	5	10	20	30	40	50	60
^{234}U	-0.8%	-0.9%	-0.9%	-1.0%	-1.0%	-1.1%	-1.1%	-1.2%	-1.2%
^{235}U	-0.61%	-0.62%	-0.63%	-0.66%	-0.76%	-0.91%	-1.09%	-1.27%	-1.42%
^{236}U		0.1%	0.0%	-0.1%	-0.2%	-0.3%	-0.4%	-0.4%	-0.5%
^{238}U	-0.069%	-0.071%	-0.074%	-0.079%	-0.081%	-0.083%	-0.085%	-0.087%	-0.089%
^{238}Pu		3.8%	3.7%	3.1%	1.5%	0.8%	0.5%	0.3%	0.1%
^{239}Pu		1.9%	1.8%	1.4%	0.4%	0.0%	-0.1%	-0.2%	-0.2%
^{240}Pu		1.9%	1.6%	1.9%	1.3%	0.8%	0.5%	0.3%	0.1%
^{241}Pu		4.4%	3.0%	1.2%	1.1%	0.6%	0.3%	0.1%	0.0%
^{242}Pu		4.8%	3.8%	2.7%	1.4%	1.2%	1.1%	0.8%	0.6%
^{237}Np		2.2%	2.1%	1.6%	0.6%	0.2%	0.0%	-0.1%	-0.2%
^{241}Am		4.5%	3.7%	2.4%	1.1%	0.7%	0.3%	0.1%	-0.1%
^{243}Am		7.3%	6.5%	4.9%	1.9%	1.5%	1.2%	1.0%	0.8%
^{103}Rh		-0.2%	-0.2%	-0.1%	0.2%	0.2%	0.1%	0.1%	0.0%
^{133}Cs		-0.3%	-0.3%	-0.2%	-0.1%	0.0%	0.0%	-0.1%	-0.1%
^{143}Nd		-0.4%	-0.4%	-0.3%	-0.2%	-0.2%	-0.2%	-0.3%	-0.3%
^{145}Nd		-0.4%	-0.3%	-0.2%	-0.1%	-0.1%	-0.1%	-0.1%	-0.1%
^{155}Gd (FP)		0.5%	1.6%	1.1%	0.5%	0.2%	-0.1%	-1.1%	-0.01%
^{95}Mo		-0.4%	-0.4%	-0.4%	-0.2%	-0.1%	-0.1%	-0.1%	-0.1%
^{99}Tc		-0.3%	-0.3%	-0.2%	-0.1%	0.0%	0.0%	-0.1%	-0.1%
^{101}Ru		-0.3%	-0.2%	-0.1%	0.0%	0.0%	0.0%	0.0%	0.0%
^{109}Ag		1.0%	1.2%	1.3%	1.1%	0.8%	0.6%	0.5%	0.3%
^{147}Sm		-0.4%	-0.5%	-0.5%	-0.4%	-0.3%	-0.3%	-0.3%	-0.3%
^{149}Sm		1.1%	1.9%	0.3%	-0.3%	-0.3%	-0.4%	-1.0%	-0.3%
^{150}Sm		-0.5%	-0.3%	0.0%	0.1%	0.1%	0.1%	0.0%	0.0%
^{151}Sm		-0.2%	0.1%	0.3%	0.0%	-0.1%	-0.2%	-0.3%	-0.1%
^{152}Sm		-0.3%	-0.3%	-0.2%	0.1%	0.1%	0.1%	0.1%	0.0%
^{153}Eu		0.0%	0.3%	0.5%	0.3%	0.3%	0.2%	0.2%	0.1%

2.4 Conclusion

The most significant effects observed on the used fuel composition involved the plutonium and americium isotopes. The increase in the concentration of these nuclides of a few percent is due to neutron spectrum hardening, which caused an energy increase in the integrated flux, together with a modification of the capture cross-section integrated into one group of the ^{238}U and the plutonium isotopes. The increase in the concentration of these nuclides is even higher due to the gadolinium in the assembly.

This increase lessened as soon as the burnup exceeded 20 GWd/MTHM, reflecting the depletion of the gadolinium 155 in the gadolinium-poisoned rods.

3. IMPACT ON THE K_{INF} OF A STORAGE POOL

Below we present the differences between the K_{inf} for an infinite lattice of UOX assemblies in the pool, irradiated with gadolinium rods, and the K_{inf} for an infinite lattice of UOX assemblies in the pool, irradiated without gadolinium rods.

The K_{inf} is calculated by taking the BUC nuclides into account, namely:

- actinides ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{237}Np, ^{241}Am and ^{243}Am;
- the 15 fission products: ^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{152}Sm, ^{155}Gd, ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{109}Ag, ^{145}Nd, ^{147}Sm, ^{150}Sm, ^{151}Sm and ^{153}Eu. With regard to the ^{155}Gd, we added the concentration of the ^{155}Gd produced by fission to the concentration of the ^{155}Gd contained in the gadolinium-poisoned rods.

In order to separate the used fuel composition problem from the "end effect" problem, the axial burnup profile was taken to be uniform and equal to the assembly mean burnup.

Table 28 shows the differences between the K_{inf} for an infinite lattice of assemblies in the pool, irradiated with gadolinium rods, and the K_{inf} for an infinite lattice of assemblies in the pool, irradiated without gadolinium rods. In this case, we do not take account of cooling times between the end of irradiation and storage in the pool.

Table 28: Observed biases in reactivity of the storage pool

	$[K_{inf}$ (UOX-Gd_2O_3) - K_{inf} (UOX)] x 10^5		
BU of assembly lattice	**4 Gd_2O_3 0.72 wt% ^{235}U**	**8 Gd_2O_3 0.72 wt% ^{235}U**	**4 Gd_2O_3 2.6 wt% ^{235}U**
2 GWd/MTHM	-2,790	-2,800	-2,560
5 GWd/MTHM	-2,550	-2,590	-2,250
10 GWd/MTHM	-1,980	-2,010	-1,740
20 GWd/MTHM	-260	-270	-145
30 GWd/MTHM	-300	-430	-220
40 GWd/MTHM	-370	-450	-240
50 GWd/MTHM	-510	-540	-180
60 GWd/MTHM	-470	-480	-350

Table 28 shows that despite the increase in the concentration of plutonium isotopes, the Kinf decreases due to the presence of gadolinium 155 in the gadolinium-poisoned rods. This decrease lessens as soon as the burnup exceed 20 GWd/MTHM, reflecting the consumption of the gadolinium in the gadolinium-poisoned rods.

However, a study showed that cooling has no effect on the impact of the UO_2-Gd_2O_3 rods.

APPENDIX 12

STUDY OF THE STORAGE OF USED FUEL ASSEMBLIES WITH GADOLINIUM

1. INITIAL COMPOSITION OF THE FUEL

For fuel assemblies with gadolinium rods, the only differences from a standard fuel assembly are the ^{235}U (or fissile Pu) enrichment of the uranium support and the Gd enrichment.

The support is studied to minimise the local power when the odd nuclides, ^{157}Gd and ^{155}Gd, disappear. For the usual assemblies with gadolinium (8 wt% Gd_2O_3), this occurs at 10,000 MWd/MTHM locally.

Due to axial leakage and the reduced density of the water in the upper part of the assembly, this burnup is obtained for a mean assembly burnup of around 25,000 MWd/MTHM.

This is the period during which the presence of Gd causes additional axial perturbation for the assembly. Therefore, the detailed examination of the axial distribution of these assemblies is imperative since it is more penalizing.

2. PARAMETERS OF THE DEPLETION OF THE FUEL

As in the case of fuel that is not poisoned with gadolinium, the final composition of the assemblies containing rods with Gd_2O_3 depends upon the thermal conditions of the rods and the moderator:

- fuel temperature;
- water temperature and density;
- power history;
- axial offset of the assembly during depletion;
- radial gradient of the power;
- presence or absence of control rods.

3. ESTIMATION OF THE AXIAL BURNUP

For the estimation of the axial burnup, the same approach was used as that adopted by AREVA-NP for the standard assemblies.

The assemblies in the core are currently monitored by calculating the scattering of neutrons together with the thermal-hydraulics. For AREVA NP – France, the axial mesh used has between 16 and 20 mesh points. This division is sufficient to monitor the variations in power without losing important information.

Thus, the axial breakdown used for the criticality calculations, after conducting various sensitivity analyses, was between 7 and 20 mesh points. The axial homogenisation at each mesh point may lead to non-conservatism if there is a small number (~ 7) mesh points, hence the incentive to use the scattering code mesh (of around 20 cm).

4. NUCLIDES

The nuclides concentrations are provided as for the assemblies without gadolinium by APOLLO2.

To the 30 nuclides (actinides and fission products) we add the depletion of the six gadolinium isotopes (^{154}Gd to ^{160}Gd). This approach corresponds to that proposed by the OECD, with a conservatism of around 6% to 10% for the negative reactivity of the used fuel, by not taking account of the minor fission products.

5. UNCERTAINTIES TAKEN INTO ACCOUNT

The uncertainties include the following:

- technological uncertainties (enrichment, geometry, mass) relating to the components of the assembly studied, and to the storage structure;
- the local burnup uncertainty, calculated by the core distribution code or measured;
- the APOLLO2 calculation of the used fuel composition;
- the spectral effect due to the presence of the rods;
- the METHOD uncertainty, representing the C/M deviation from the MC code used in relation to the critical experiments on which it was validated;
- the statistical uncertainty of the MC code.

For the assemblies with Gd, the additional uncertainty is related to the concentration of residual odd nuclides of Gd in the case of fairly low assembly burnups (<25,000 MWd/MTHM). Beyond this limit, only the even nuclides of Gd influence the k_{eff} and, in this case, the uncertainty is very low given the residual effectiveness of these nuclides.

6. CRITICALITY CALCULATION

The lattice studied is generally conservative based on the part of the pool taken into account in the MC simulation. The most penalizing is obviously the infinite lattice of cells. An intermediate configuration is used at AREVA NP, with the simulation of the largest module consisting of around 100 cells. The gain is due to the fact that the inter-module water level is taken into account. The pool configuration with regions I and II is also modelled to study the impact of the interface between the two storage regions.

An important parameter for the reliability of the k_{eff} obtained is the stabilisation of the fission sources in the volume studied. This leads to very long calculations, such as in the case of an "entire pool" simulation.

7. EXAMPLES OF UO_2 WITH GADOLINIUM RODS

The minimum mean burnup observed in the balanced cycle is 37.7 GWd/MTHM for an assembly with eight gadolinium rods and 38.5 GWd/MTHM for an assembly without gadolinium rods.

An initial analysis is carried out to compare the 2D and 3D approaches.

Table 29 below shows that this burnup is at the limit with regard to the conservatism of the 2D method, thus without axial leakage. This leakage is found to have an impact on k_{eff} of about 300 pcm. Taking the axial burnup into account results in perfect equality (k_{eff} = 0.9400).

The same burnup but with eight gadolinium rods and a new axial burnup leads to a slight decrease in reactivity. This configuration is equivalent, in terms of the k_{eff} value, to an assembly at 38.5 GWd/MTHM without gadolinium, studied in 3D with its own axial burnup profile.

Table 29: k_{eff} for UO_2 with 4.45 wt% ^{235}U

Case	Mean burnup	Axial burnup	Model	Number of gadolinium rods	K_{eff}
1	37.7 GWd/MTHM	No	2D	0	0.94
2	37.7 GWd/MTHM	No	3D	0	0.93
3	37.7 GWd/MTHM	Yes	3D	0	0.94
4	37.7 GWd/MTHM	Yes	3D	8	0.93
5	38.5 GWd/MTHM	Yes	3D	0	0.93

These results were obtained for a burnup characteristic of a 1/3 management method and an ^{235}U enrichment limited to 4.45 wt%.

This situation is very different for EPR management methods with 5.0 wt% ^{235}U and for which assemblies with 24 gadolinium rods are frequently used. The axial profile is modified along with the residual penalty for the even nuclides. For these assemblies, we therefore expect to see significant differences between the five cases shown in the table above.

Table 30 below shows the differences between the axial profiles for three assemblies containing 0, 8 and 24 gadolinium rods. The profile with 24 gadolinium rods displays low values in the upper and lower parts, which should result in a significant difference in reactivity between the three configurations with the same mean burnup.

Table 30: Axial burnup profile for UO_2 with 5.0 wt% ^{235}U

Axial mesh	UO_2 - 0 Gd	UO_2 - 8 Gd	UO_2 - 24 Gd
18 (upper)	0.626	0.613	0.583
17	0.891	0.894	0.886
16	0.967	0.996	1.008
Central mesh points	1.079	1.076	1.080
3	1.071	1.069	1.060
2	1.000	0.989	0.965
1 (lower)	0.726	0.700	0.654

8. ORDERS OF MAGNITUDE

Here are some of the typical values for the storage of used fuel assemblies with gadolinium currently used:

- residual negative reactivity of the even gadolinium isotopes ~ -75 pcm / Gd rod;
- axial burnup penalty between 1,000 and 2,000 pcm for a high mean burnup (> 45 GWd/MTHM);
- spectral effect of the control rods ~ +300 pcm;
- enrichment uncertainty ~ +800 pcm / +0.1 wt% U235 (5.0 wt% to 5.1 wt%);
- region II storage lattice pitch ~ 500 pcm /mm (lattice 240cm);

- calculated mean burnup ~ 4%;
- nuclide concentration calculated with APOLLO2 ~ 500 to 1,000 pcm;
- uncertainty method ~ 1,100 pcm.

9. SUMMARY

No specific differences are observed between the storage of the used fuel assemblies with gadolinium and standard storage in region II.

The uncertainties used for the storage of fuel without gadolinium are therefore renewed.

The residual even gadolinium nuclides has a limited effect and the impact of their uncertainties (cross-sections - concentrations) on the k_{eff} is low.

However, particular care is required for the axial modelling of the fuel.

The axial burnup profile is dependent on the mean burnup and the number of gadolinium rods in the studied assembly. This parameter can generate great differences in reactivity, despite the residual presence of the even gadolinium nuclides: at equal mean burnup, the effect of the axial burnup can prevail over the residual penalty of the even gadolinium nuclides.

The used fuel with gadolinium must therefore be represented with the maximum number of axial meshes possible (> 10).

APPENDIX 13

SIMULATION METHODS AVAILABLE IN THE MORET 4 CODE

The **natural** or conventional or analogous (to reality) method: this is the method that is usually available and used in the Monte Carlo criticality codes; the number of neutrons produced in a volume, divided by the total number of neutrons produced in the system, represents the proportion of neutrons to be emitted from this volume during the next stage.

The **stratified** method: this does not allow the number of source neutrons to be generated in a fissile volume to be zero when the number of neutrons, sources potentially emitted in this volume (Pv), is less than 0.5. (Production in this volume is divided by the sum of the system entire production.) In this case, (Pv<0.5) with the previous method, no neutrons would be generated in the volume in question, whereas with the stratified method, a source neutron is generated with a weight equal to the actual number of source neutrons; the aim of this strategy is to guarantee that every fissile zone (that has very little interaction with the rest of the system) contains, with each generation, at least one source neutron with a weight possibly less than 1.

The **kij** method: this is a simulation option that uses the eigenvector associated with the highest eigenvalue in the kij matrix to estimate the distribution of the source neutrons in the following stage, with the aim of accelerating the convergence of the sources; this method is very similar to the stratified method, apart from the use of the eigenvector associated with the highest eigenvalue in the kij matrix to renormalize the distribution of the source neutrons at every stage "Freq" (Freq is an integer specified by the user); the kij matrix gives the number of neutrons emitted by fission in the volume i from a neutron emitted in the volume j; the highest eigenvalue in this matrix is equal to the system multiplication factor, and the associated eigenvector corresponds to the distribution of the neutrons in the different volumes.

The "**super-history**" method: this is a simulation option used to monitor a source neutron and its descendants over L generations in order to avoid recalculating the distribution of the sources for each generation; for all methods other than the "super-history" method, the number of neutrons simulated with each generation is constant; it was shown that this normalization was the cause of a bias on the calculated k_{eff} value and its variance; this was because by considering a constant number of neutrons for each generation, smaller zones were likely to be favoured; this method allows the population of neutrons to vary over L generations; normalization is only performed after these L generations.

APPENDIX 14

UNCERTAINTIES ASSOCIATED WITH FUEL ASSEMBLY BURNUP MEASUREMENTS

I PURPOSE

The purpose of this appendix is to summarise the major tendencies in terms of uncertainties associated with a burnup measurement, depending on the measuring devices used and the types of assemblies measured.

This appendix summarises the various physical principles on which the various burnup measurement methods and devices are based, before addressing the uncertainties associated with these methods.

II PHYSICAL PRINCIPLES FOR BURNUP MEASUREMENT

Measuring burnup may require the use of two different types of measurements:

- gamma spectrometry;
- passive neutron measurement.

These two methods are associated with gamma spectrometry measurement: the absolute measurement of ^{137}Cs and the Cs ratio method (^{134}Cs/^{137}Cs).

1) Absolute measurement of ^{137}Cs

This radionuclide is a direct fission product. It has no precursor and has a small capture cross-section. Due to the link between the burnup and the number of fissions, the ^{137}Cs is produced proportionally to the burnup and decays according to its half-life. Furthermore, its thermal fission yields for both ^{235}U and ^{239}Pu are identical. Therefore, the quantity of ^{137}Cs (or its activity) is independent of the initial quantity of ^{235}U, and thus the enrichment. Finally, it is not sensitive to variations in the irradiation history (IH) due to its long period of decay, of around 30 years.

Its mass or its gamma activity, expressed as Cs137, takes the following form:

$$\boxed{Cs137 = \alpha\, BU\, \exp(-\lambda_{Cs137}\, CT)}$$

where: α is a coefficient of proportionality
BU represents the burnup
λ_{Cs137} denotes the radioactive decay constant of the ^{137}Cs
CT represents the cooling time

The method is therefore a measurement of the 662 keV ray of the ^{137}Cs, which is directly proportional to the burnup.

2) Cs ratio method (^{134}Cs/^{137}Cs)

The main production route of ^{134}Cs is shown in the figure below:

$$fission \longrightarrow {}^{133}I \xrightarrow[20.8hrs]{} {}^{133}Xe \xrightarrow[5.2days]{} {}^{133}_{stable}Cs \xrightarrow{(n,\gamma)} {}^{134}Cs$$

Figure 21: Production of ^{134}Cs

It can be seen that this production requires two neutron interactions. Therefore, unlike the ^{137}Cs, the fact that the ^{134}Cs needs two neutrons to be produced means that it develops proportionaly to the square of the burnup. Moreover, this nuclide decays according to its approximately two-year half-life and it is independent of the enrichment for the same reasons as the ^{137}Cs. However, due to its short half-life and the length of the fuel cycles (around one year), this radionuclide is sensitive to the irradiation history (IH).

Therefore, its mass or gamma activity, expressed as ^{134}Cs, follows the law below:

$$Cs134 = \beta\, BU^2 \exp(-\lambda_{Cs134}\, CT)\; f(IH)$$

where: β is a coefficient of proportionality
BU represents the burnup
λ_{Cs134} denotes the radioactive decay constant of the ^{134}Cs
CT represents the cooling time
f is a function of the irradiation history (IH)

The cesium activity ratio (^{134}Cs/^{137}Cs) thus follows a law that is proportional to the burnup but dependent on the irradiation history. This method therefore requires to know the fuel irradiation history (the mean power or burnup for each irradiation cycle and the dates of these cycles).

3) Neutron method

First of all, neutron emission is mainly caused by the production of curium 244 for standard burnups for assemblies irradiated in light water reactors (however for very low burnups of UNGG type reactors, plutonium is the main neutron emitter). For light water reactors, ^{244}Cm is the predominant neutron emitter and decays according to its half-life of around 18.1 years. This makes it insensitive to the irradiation history but it has a

dependency according to the burnup, based on the nature of the fuel. This is because for a UOX fuel, ^{244}Cm will be produced from the ^{238}U after a long capture chain.

For MOX fuels, the capture chain starts predominantly from the ^{239}Pu, a nuclide located at a shorter "relationship distance" from the ^{244}Cm, involving fewer neutron reactions. Therefore, while the total neutron emission for a UOX depends on BU^{α} where α varies between 4 and 5, the total neutron emission for a MOX depends on BU^2.

Furthermore, for a UOX, if the ^{235}U enrichment increases, then less ^{239}Pu is produced, as it comes mainly from the capture of ^{238}U, which is in a smaller quantity. This leads to a decrease in the quantity of ^{244}Cm produced. This highlights the dependence of neutron emission on enrichment for a UOX fuel. As far as MOX fuels are concerned, due to the link between the initial quantity of Pu and the production of ^{244}Cm, the neutron emission has a dependence on the Pu content, expressed as c_Pu.

This measurement requires the use of an actinide depletion calculation code for actinides, ranging from ^{235}U to ^{244}Cm. This is the reason why this method was the latest to be used in industry.

The neutron emission (EN), linked to the production of the radionuclide ^{244}Cm, and depending on the nature of the fuel, is expressed in the following two forms:

For a UOX fuel

$$EN_{UOX} = \kappa \, BU^{\alpha} \exp(-\lambda_{Cm244} \, CT) \, g(ENR)$$

where: κ is a coefficient of proportionality
BU represents the burnup
α varies between 4 and 5
λ_{Cm244} denotes the radioactive decay constant of the ^{244}Cm
CT represents the cooling time
g is a function of the enrichment (ENR)

For a MOX fuel

$$EN_{MOX} = \varepsilon \, BU^{2} \exp(-\lambda_{Cm244} \, CT) \, h(c\_Pu)$$

where: ε is a coefficient of proportionality
BU represents the burnup
λ_{Cm244} denotes the radioactive decay constant of the ^{244}Cm
CT represents the cooling time
h is a function of the plutonium content

Neutron emission is therefore a power function of the burnup and depends on the initial composition of the fuel.

III MEASUREMENT METHODS

1) Ionization chamber

An ionization chamber can be used to count all of the gammas emitted by a fuel assembly. When the fuel cooling time is longer than a few months, the number of gammas emitted is generally proportional to the burnup. With this type of measurement, a method similar to the absolute measurement of ^{137}Cs can be used, but with a greater level of uncertainty due to the incomplete linearity of the method. This detector is used to track the burnup profile in the PYTHON system.

2) CZT gamma spectrometry

A cadmium telluride (Cd-Zn-Te) detector is used to perform a gamma spectrometry analysis with a medium resolution (peak width of around 2%). This type of semi-conductor detector has the advantage of being able to operate at ambient temperature and accurately extracts the ^{134}Cs peak at 796 keV and the ^{137}Cs peak at 662 keV. However, the measurement uncertainty will be slightly higher than with a germanium detector. This detector is used in the SMOPY system.

3) Gamma spectrometry with a germanium detector

The germanium detector operates at -200°C and therefore requires a cooling system. However, it has an excellent resolution (<0.2%). This type of detector is used on the permanently installed systems in workshops T1 and R1 at the La Hague plant. With this system, the counting uncertainties are negligible.

4) Fission chamber

The fission chamber has an enriched uranium deposit. The thermal neutrons create fissions in the chamber and the resulting fission products ionize the gas inside it, generating an associated count rate.

If this method is used in water, the neutrons are thermalized and a bare fission chamber can be used. At the La Hague plant, measurements are taken in an air environment; the fission chambers are therefore surrounded by polythene to thermalize the neutron flux exiting the assembly.

IV UNCERTAINTIES OF THE VARIOUS METHODS AND MEASURING SYSTEMS

The measuring systems installed in the T1 and R1 workshops at the La Hague plant include gamma spectrometry and fission chamber neutron measurement. This is the most effective system with the minimum level of uncertainty. First details of this system are provided, then the sources of uncertainty associated with the other measuring systems are described.

1) Measuring system installed at La Hague

Thanks to operation feedback, we were able to compare the expected neutron activities and emissions with the measurements. These values are shown in the table below.

	Cs134 uncertainty	Cs137 uncertainty	Neutron uncertainty
Minimum	-6.9%	-5.1%	-12.7%
Maximum	8.8%	8.1%	34.7%
Mean	-0.2%	0.8%	1.3%
2 sigma	**6.1%**	**4.7%**	**13.4%**

The observed discrepancies combine three different aspects as follows:

- the measurement uncertainties;
- the uncertainties related to the values provided by the reactor (irradiation history);
- uncertainties related to the use of the CESAR code to evaluate the correlation functions between a measured parameter and the burnup.

For the ^{137}Cs, the measurement uncertainties and the uncertainty relating to the use of CESAR are negligible.

2) Minimum uncertainties associated with a measuring system

Under measuring conditions, the uncertainty inherent to the measurement calibration coefficient must also be added to the uncertainties discussed in the paragraph above. This additional uncertainty has a minimum value of 5%. **Thus for gamma spectrometry,** the total uncertainty value is **around 10%.**

For neutron measurement, the total uncertainty is therefore **around 20%.**

3) Additional tendencies depending on the measurement method

a) Gamma spectrometry

The ^{137}Cs method is an absolute method and is therefore sensitive to the measuring parameters. For example, the assembly moves in front of the detector at a certain speed. The uncertainty associated with this speed has a direct impact on the measurement.

Similarly, if the assembly is repositioned, this also affects the measurement. However, these parameters do not affect the Cs ratio method as there is a compensation phenomenon. **Ultimately, the Cs ratio method is more accurate.**

If a CZT detector is used, the lack of precision associated with the evaluation of the net areas must also be added, amounting to a few additional percent.

The precision of the absolute ^{137}Cs method can, therefore, be estimated at around 15% to 25%, depending on the type of detector and the mechanical precision of the measuring system (speed, positioning, acquisition time, etc.).

The ^{134}Cs/^{137}Cs ratio method is a little more accurate, from 10% to 15%, depending on the type of detector and the acquisition time.

b) Neutron measurement

Like the ^{137}Cs method, neutron measurement is also an absolute approach.

4) Tendencies based on the types of assemblies measured

a) ^{137}Cs method

The ^{137}Cs method is independent of the irradiation history, and the production of ^{137}Cs is linear with the burnup. The ^{137}Cs half-life of 30 years also makes the level of uncertainty stable, even for long cooling times (if this duration is known). The uncertainty of this method is relatively stable, based on variations in the assembly characteristics.

b) Cs ratio method (^{134}Cs/^{137}Cs)

The ^{134}Cs is produced by double neutron capture and, as a result, there is very little ^{134}Cs with a low burnup. The uncertainty of this method therefore increases at low burnup (when the assembly has been through only one irradiation cycle).

The ^{134}Cs half-life of two years makes the method inapplicable when the cooling times are very long, and particularly when the burnup is low. However, at the La Hague plant, for

standard burnups (three to four reactor cycles), measuring an assembly with a cooling time of 20 years is still possible.

c) Neutron measurement

Assemblies with a low burnup have an even higher impact on neutron measurement uncertainties due to the production of a very small amount of ^{244}Cm at the start of irradiation. The neutron method is only applicable to assemblies with a burnup above 10,000 MWd/MTHM. The higher the burnup, the lower the uncertainty of this method.

Compared to the gamma measurement method, this approach also has the advantage of being representative of the entire assembly. This is because with gamma spectrometry, the measured energies are lower than 800 keV and, consequently, the measuring system sees only the first three rows of rods.