Burnup credit implementation for transport and storage casks of irradiated UO_2 fuel assemblies

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ABSTRACT

TN International currently uses the BUC method for the design of casks dedicated to the transport of PWR UO_2 used fuel assemblies. As long as the fuel enrichment of the PWR fuel assemblies was sufficiently low, a simplified BUC method based on the sole consideration of 8 major actinides and the use of a simplified axial burnup approach was satisfactory to cover the needs without necessity to design new casks.

Nevertheless, the continuous increase of the fuel enrichment during the last decade has led TN International to continue the investigations on an improved BUC method in order to limit both the increase of the neutron poison content in the new basket designs and the burnup constraints attached to the acceptability of the fuel assemblies for transport. The strategy of TN International was then to take benefit of the reactivity reserves, which might be gained by considering the main fission products (^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm,^{152}Sm and ^{155}Gd) that make up 50% of the negative reactivity of all fission products and more realistic axial profiles of burnup instead of uniform axial burnup profiles .

The "BUC" calculation route for PWR UO_2 used fuel is based on the connection of the French depletion code DARWIN and the French Criticality Safety Package CRISTAL V1 which are developed by the CEA and the IRSN in collaboration with French nuclear industry. French BUC experimental programs have been separately performed in Cadarache (France) and in Valduc (France) in order to validate respectively the DARWIN depletion code and the CRISTAL V1 Criticality Safety code system.

The aim of this article is to present the improved BUC method implemented at TN International for the criticality safety assessment of transport and storage casks.

1. Introduction

In the early 1980's, TN International has used a burnup credit (BUC) calculation method, taking only major actinides into account, for carrying out the criticality safety analysis related to the transportation of PWR UO_2 used fuel assemblies. This method is based on the use of very conservative assumptions: only major actinides are taken into account and the burnup along the active length of the assembly is supposed to be equal to the mean burnup in the 50 least-irradiated centimetres of the assembly. In order to respond to the increase of PWR UO_2 fuel initial enrichment, TN International has made an effort for the implementation of the BUC method including not only the major actinides but also the fission products. However, the utilization of BUC for irradiated fuel assemblies requires the definition of a method that ensures the conservatism of calculations and the experimental validation of the used fuel inventory for actinides and Fission Products. In addition, BUC requires an accurate knowledge of the isotope cross-sections (actinides and fission products).

2. Current burnup credit method

The actinides-only BUC method is based on the use of the following very pessimistic assumptions:

- 8 major actinides (^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu)
- Specific power : 40W/g
- One irradiation cycle
- No cooling time
- The mean burnup averaged over the least irradiated 50 cm of the fuel's active length is taken into account on the entire fuel's active length.

The calculation procedure (see Figure 1) used by TN International for the actinides-only BUC method is based on:

- Depletion calculation for the actinides is done using DARWIN [1].
- k_{eff} calculation of the loaded transport cask carried out by using the French Criticality-Safety Package CRISTAL V1 [2].

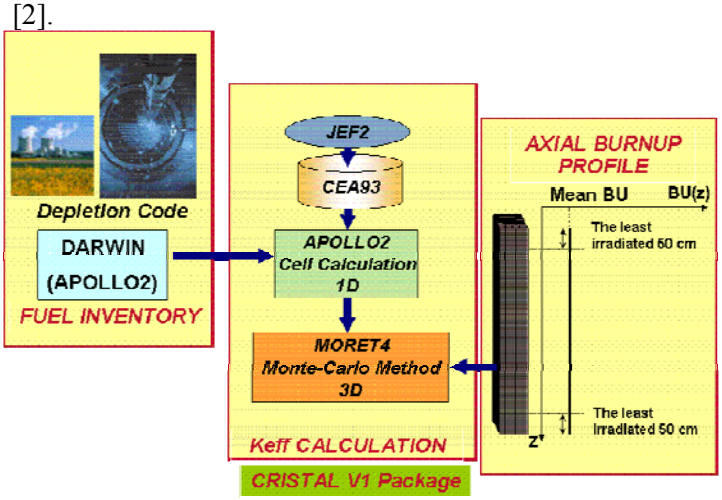

***Fig 1**: Calculation procedure for actinides-only BUC method*

In the previous method the burnup verification procedure depends on the magnitude of burnup guaranteed on the top and bottom 50 cm of the fuel active part as follows:

− BU ≤ 3 200 MWd/tU ➔ Qualitative burnup verification (irradiation check),
− BU > 3 200 MWd/tU ➔ Quantitative burnup verification (burnup measurement).

The actinides-only method was accepted by the French safety authorities and this method has been used for more than 25 years by TN International on the TN12, TN13 and TN17 transport cask family used for the transportation of PWR uranium oxide used fuel assemblies from the European reactors to the nuclear reprocessing plant of AREVA NC in La Hague (France). Nevertheless, this method is quite limited today as it does not allow taking profit from the negative reactivity worth of the fission products and a more realistic axial burnup profile.

3. Advanced burnup credit calculation method

Since the initial enrichment of fuel assemblies is increasing from 4% to 5%, it is needed for transportation and storage of used fuel to reduce the conservatism due to the very pessimistic assumptions of the actinides-only method. For that purpose, a working group was formed in 1997, gathering most of the French nuclear companies and IRSN (technical support of the safety authorities), to analyze different propositions to introduce some fission products plus a more realistic axial profile of burnup into the criticality studies.

In the last years TN International has implemented a new improved BUC calculation method taking into account actinides and fission products. From the 12 actinides and 15 fission products recommended by the OECD [3], the following actinides and fission products are considered in the criticality calculation:

− 9 actinides: ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{241}Am,

− 6 fission products: ^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, 1^{52}Sm, ^{155}Gd.

The limitation to 9 actinides and 6 fission products used in the improved BUC method implemented in the criticality calculation route is conservative compared to the OECD recommendation and represents approximately 50% of the total worth of all fission products.

The improved BUC method further deals with:

- the definition of **conservative conditions of irradiation**, in order to guarantee the conservatism of the used fuel inventory,
- the **experimental validation of the used fuel inventory** and the reactivity worth of actinides and FPs used in the BUC
- the definition of **bounding axial burnup profile evaluated** from reactor data or burnup measurements,
- the **validation of the criticality code** and nuclear data associated.

The CEA and the IRSN are developing in collaboration with French nuclear industry a BUC calculation route for PWR-UO_2 assemblies [4], based on the connection of the French depletion code DARWIN [1] and the French Criticality Safety Package, CRISTAL V1 [2].

The DARWIN depletion code calculates burnup depending concentrations of isotopes during irradiation and after the cooling time. These concentrations, to which a correction factors can be applied, are used then as input data in the Criticality Safety Package CRISTAL V1, which provides the neutron multiplication factor associated to the calculated configuration. The sequence of the codes is presented hereafter:

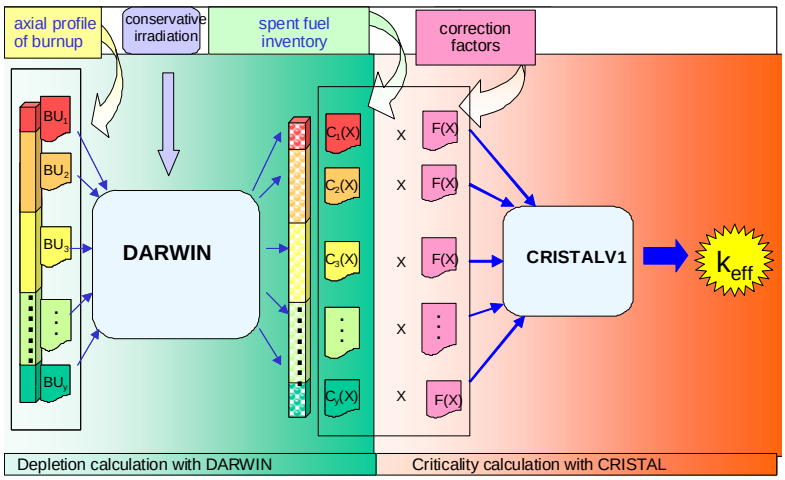

Figure 2: *Presentation of the French "BUC" calculation route*

3.1. Conservative approach for depletion calculation

The parameters relevant for the depletion calculation are: specific power, fuel temperature, moderator temperature, moderator density, boron concentration in the moderator, irradiation history, fuel assembly environment during irradiation, control rod insertion, burnable absorbers as part of the fresh fuel and cooling time.

The reactivity of burned fuel increases when the irradiation conditions lead to a hardening neutron spectrum, this hardening being dependent on control rod insertion, MOX environment and irradiation parameters.

- ***Fuel specific power***

The specific power used in the depletion calculations has a slight effect on the reactivity in the 30-50 W/g PWR operating range [4]. The value of the conservative specific power used in the depletion calculation is thus 40 W/g, with no inter-cycle downtime in order to minimize ^{241}Pu decay.

- ***Fuel temperature***

The maximum effective mean fuel temperature in the core can be considered as it induces neutron spectrum hardening. Indeed, it is conservative to consider a high value for the fuel temperature as it leads to more resonant captures on ^{238}U, and then to further production of ^{239}Pu. However, it should be noticed that the fuel temperature used in the depletion calculation has a small effect on the criticality calculation, about +5 pcm/°C at 50 GWd/t_{HM} [6]

- ***Moderator temperature***

The moderator temperature increases with the fuel height in the core (i.e. the moderator temperature is higher at the top of the core than at the bottom). For PWR assemblies, when the moderator temperature increases, the moderator density decreases, which thus leads to a reduction of the moderation and, therefore, to spectral hardening. So, it is conservative to consider in the depletion calculation the mean of the moderator temperature corresponding to the mean temperature of the outlet water and apply it to the full length of the assembly. Studies performed in [6] show an increase of the reactivity, about +80 pcm/°C at 50 GWd/t_{HM}.

- ***Natural boron concentration***

Increasing boron concentration results in spectral hardening due to the absorption of thermal neutrons in the moderator by boron. So, it is conservative to consider a high value of the boron concentration as it leads to an increase in reactivity.

A soluble boron concentration of 1500 ppm is typical of Beginning-Of-Cycle (BOC). This value decreases down to 900 ppm during the first days due to Xenon and Samarium poisoning, and afterwards decreases linearly down to 0 ppm at the End-Of-Cycle. A conservative value of 800 ppm can be retained according to [7]. This value corresponds to an average boron concentration of a full cycle.

- ***Environment of the depleted UO_2 assembly***

The depletion of the UO_2 assembly is performed considering the UO_2 assembly surrounded by 8 MOX assemblies at 15 GWd/t_{HM} [5] during the entire irradiation (see Figure 3). This is a conservative approach because, in France, only twenty 900 MWe PWRs are currently devoted to recycling Pu in 30% mixed core loading. The choice of 8 MOX assemblies is justified by the fact that AREVA NC stores and transports assemblies coming from not only French reactors but also assemblies coming from foreign reactors which can contain more than 30% of MOX assemblies in the core.

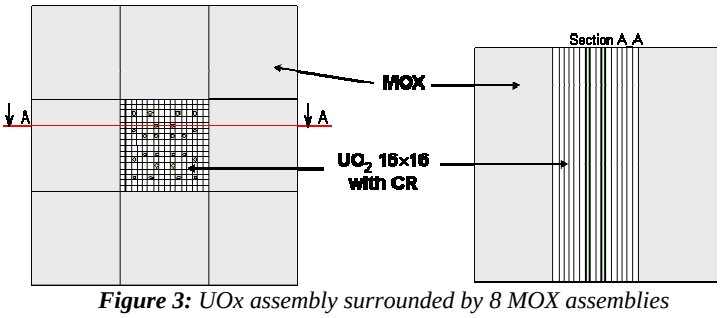

Figure 3: *UOx assembly surrounded by 8 MOX assemblies*

- ***Control rod insertion***

Reactors operations can involve periods of partial control rod (CR) insertion. In order to maximise the reactivity effect due to this insertion, the effect of a full axial control rod insertion (Ag-In-Cd or B_4C) during the entire burnup can be considered. The reactivity of a transport and storage cask loaded with UOx assemblies irradiated with CRs fully axially inserted from 0 to 40 GWd/t is about 4000 pcm higher than the reactivity without CRs. These 4000 pcm include two large conservatisms: the time of CR insertion (all the

irradiation) and the level of axial insertion of CRs (full axial insertion). But although the probability is low that an assembly is irradiated with CRs inserted during all its irradiation, it is difficult to exclude that possibility. Furthermore, it is possible to consider a partial axial CR insertion provided that the operator guarantees this limited insertion of the CRs.

- ***Cooling time***

Many studies have shown [9] that, after the irradiation and up to a cooling time of 100 years, the reactivity decreases. This decrease is mainly due to the decay of the ^{241}Pu (to ^{241}Am) plus the increase of ^{155}Gd. After 100 years of cooling time, the reactivity starts to increase again (as the ^{241}Am decays) until around 30 000 years. In terms of cooling time, the need for the transport and storage casks is a cooling times of less than 50 years. For transport and storage casks' applications, it should be acceptable to consider in the criticality studies the minimum of the cooling time that can be justified by the operators.

3.2. Experimental validation of the used fuel inventory

The BUC implementation for a transport cask application including actinides and six fission products is performed at TN International using DARWIN, the French BUC calculation route with its associated libraries based on JEF2.2. The BUC method includes the validation of the code systems dedicated to the calculation of the used fuel inventory and the definition of a set of isotopic correction factors (CF) used in the criticality calculations. A complementary way is the implementation of k_{eff} penalty terms, derived from PIE and reactivity worth measurements, in the k_{eff} acceptance criterion as described in [21].

The determination of the isotopic correction factors or k_{eff} penalty terms is a consequence of the experimental validation of the DARWIN depletion code in order to guarantee a conservative k_{eff}.

A BUC program [8] has been developed at Cadarache Centre in the framework of the CEA-AREVA collaboration in order to validate fuel inventory calculations. This program involved chemical analyses and microprobe measurements of PWR fuel rods to obtain the fuel inventory (Actinides and FPs). This experimental datas are based on chemical analysis measurements from fuel rod cuts irradiated in French PWR reactors. This enables to cover a large range of UO_2 fuels with various enrichments in ^{235}U, from 3.1% to 4.5%, and burnups from 10 GWd/t to 60 GWd/t.t.

3.3. Bounding axial and horizontal burnup profile

The burnup distribution in the fuel assembly can have significant impact on k_{eff} and needs to be assessed. Indeed, the criticality of the irradiated fuel assemblies is affected by their axial burnup shapes. The fuel assembly burnup has axial and horizontal gradients due to the neutron flux distribution during irradiation. This flux shape is mainly related to:

- Neutron leakage at the top and the bottom of the assembly and if considered, also the presence of partially inserted control rods.
- Radial neutron leakage (caused by the environment of the assembly and its position in the reactor during irradiation).

The effect of the assembly axial burnup profile in burnup credit calculations was investigated in the OECD/NEA benchmarks Phase IIA [18], IIB [19] and IIC [20]. Due to a strong shift of the flux towards the fuel pin extremities with increasing burnup, the simplified flat burnup model is not conservative for burnup credit calculations of PWR casks and the storage of highly irradiated assemblies, i.e. mean BU > 30 GWd/t.

A more realistic but still bounding axial profile can be defined for the criticality studies by:

- **Examining profiles come from measurements**. In France, more than 3000 assemblies irradiated in French reactors have been measured at La Hague – AREVA NC reprocessing plant and were examined. These measurements pointed out that most of the assemblies coming from French reactors have quite similar

profiles (due to the operating conditions). One method to show the conservatism of the axial bounding profile generated from measurements is to verify on each point of the bounding profile that the burnup value is lower than the real measured profile value. The Figure 4 below gives an example of a determination.

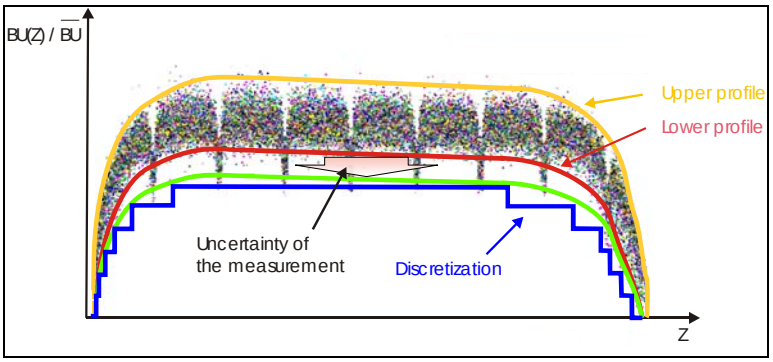

Figure .4: *Example of a conservative axial profile determination*

- **Calculating different types of profiles based on reactor records** and determining a penalizing one. The method to generate a bounding axial profile on the basis of representative axial fuel assembly burnup shapes of nuclear plants which derived from in-core 3D power density distribution measurements is described in [10].

However, the conservatism of the axial profile used in the criticality studies will have to be demonstrated for both the methods used for determining the bounding axial profile. Therefore:

- If the profiles have been calculated, the conservatism of this calculation have to be guaranteed.
- If the profiles have been determined by measurements, the uncertainties due to the method used for measurement have to be determined.

In addition, particular care should be taken when control rods can be partially inserted in the fuel assemblies.

The effect of a potential horizontal burnup variation on the reactivity of the fuel assemblies has also to be considered in the criticality safety analysis. Some calculations have been carried out giving the gradients as a function of burnup [11]. Measurements made at La Hague gave indications of values of horizontal gradients as a function of burnup [12] for a large amount of fuel assemblies. Depending on these studied cases this horizontal effect may have to be considered, for example with:

- a deterministic approach considering the less-irradiated faces close together,
- or a probabilistic approach.

3.4. Validation of the criticality code

Studies considering fuel burnup require an accurate knowledge of the cross-sections of isotopes (actinides and fission products) that are not commonly used in criticality calculation when a fresh-fuel assumption is made. The CRISTAL V1 system has been validated on a large database of benchmark experiments (about 1900) [13], which includes all the different kinds of configurations encountered in the industrial nuclear fuel cycle and burnup credit configurations. Concerning this last point, a French program has been performed in order to support the development of a technical basis for burnup credit validation in the case of industrial configuration (transport, storage, reprocessing of used fuel assemblies). This program is based on two types of experimental data [14]:

- A series of critical experiments referred to as “Haut Taux de Combustion” (HTC) [15] and Fission Products (FP) [16] experimental program were performed in France at the Commissariat à l’Energie Atomique (CEA) at the Valduc research facility. These experiments were designed by Institut de Radioprotection et de Sûreté Nucléaire (IRSN) and funded by AREVA NC and IRSN. The aim of this experimental program was to validate the cross sections of major actinides and fission products.

▪ Reactivity worth measurements of the various BUC nuclides (Actinides and Fission products) by oscillation of specific fuel rod samples in the MINERVE reactor [17]. The goal of the reactivity worth measurements by the oscillation technique in the MINERVE reactor is to qualify the CRISTAL V1 calculation system and nuclear cross section data used to predict the poisoning reactivity worth of individual BUC isotopes as well as to determine the integral reactivity worth of real irradiated fuel rod samples from French reactors [22].

4. Application and gain estimation

The burnup credit reactivity versus burnup has been evaluated for a transport and storage cask loaded with 7 used fuel assemblies PWR-UO_2 type 17×17 using the "BUC" calculation route for PWR UOx used fuel is based on the connection of the French depletion code DARWIN 2.1.1 and the French Criticality Safety Package, CRISTAL V1.0 The results of the fresh-fuel assumption are compared to the burnup credit method termed "Actinides-only" "Actinides + 6 FPs" and "Actinides + 15 FPs". The comparisons between the k_{eff} value obtained with the fresh fuel assumption and the different burnup credit method could be presented as:

$$\Delta k = k_{eff}(\text{fresh fuel}) - k_{eff}(\text{burnup credit method used})$$

The calculations are performed with a uniform average burnup distribution. Some conservative correction factors are used for the fission products and for the actinides to be taken into account:

- the discrepancies observed between the results of the depletion calculations and the experimental data,
- the uncertainties in the cross sections deduced from the MINERVE experimental program.

Moreover, the conservative assumptions, regarding the conditions of irradiation described in paragraph 3.1. were taken into account. The parameters and conditions of irradiation used for the depletion calculations are summarized in the table below:

Fuel temperature (K)	923
Clad temperature (K)	673
Moderator temperature (K)	598
Specific Power (MW/t)	40
Moderator boron concentration (ppm)	800
Environment	8 MOX assemblies
Cooling time (year)	1
Control rods	24 AIC
Insertion of CRs	Throughout all the irradiation

Table 1: *Parameters and conditions of irradiation used for the depletion calculations*

In the BUC "actinides-only" method the fuel inventory in criticality calculation is composed of the following isotopes ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu.

In the BUC "actinides + FPs" method the fuel inventory in criticality calculation considers the following:

- Actinides : ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu and ^{241}Am,
- 6 FPs: ^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{152}Sm and ^{155}Gd,
- 15 FPs: ^{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.

The values of Δk for a transport and storage cask loaded with 7 used fuel assemblies PWR UO_2 type 17×17 with a maximal initial ^{235}U enrichment of 5 wt. %, are given below:

Burnup GWd/t_{HM}	"Actinide-Only" BUC	"Actinides + 6FPs" BUC	"Actinides + 15 FPs" BUC
10	-2.9 %	-5.5 %	-6.1 %
20	- 6.1 %	-9.1 %	-10.5 %
30	-8.4 %	-12 %	-14 %
40	-10.3 %	-15.2 %	-17.3 %
50	-12.5 %	-17.7 %	-20.2 %

Table 2: *Comparison between fresh-fuel assumption and different burnup credit method*

Finally, we can notice that, even if the correction factors used in the criticality calculations are determined in a conservative manner and the irradiation history is conservatively used in the depletion calculations, the improved new method based on "Actinides + 6 FPs" gives a gain up to Δk = 5.5 % for a burnup of 10 GWd/ t_{HM} to Δk = 17.7 % for a burnup of 50 GWd/t_{HM}.

5. Conclusion

The new advanced burnup credit method implemented by TN International, based on the consideration of actinides and fission products as well as on bounding axial burnup profiles, allows to extend burnup credit advantages to new transport and storage cask design developments dedicated to PWR fuel assemblies.

The new method allows an adequate level of criticality safety of the transport and storage cask designs whereas performances of the cask designs are continuously increased with high-enriched PWR fuel assemblies. This method, implemented for transport and storage cask design gives evidence that the extended burnup credit method based on fission products is adequate for high-enriched PWR uranium oxide fuel assemblies.

The calculation codes DARWIN 2 and CRISTAL V1 associated to the advanced burnup credit method have been validated for burnup credit application on a large experimental program led in France. The validation of the code system has been implemented against post-irradiation examination (PIE) data and critical benchmark experiments representative of the transport and storage cask configurations.

Taking profit of the feedback received by investigations led on different burnup credit approaches, TN International is expecting new perspectives for the transport and cask design developments by:

- Extension of the advanced burnup credit method to selected transport and storage package designs dedicated to UO_2 PWR used fuel assemblies with the advantages of:
  - reduction of burnup requirements,
  - reduction of basket poison content,
  - simplification of the burnup verification procedure.
- − Extension of the application area of the advanced burnup credit method to:
  - MOX PWR fuel assembly,
  - UO_2 BWR fuel assembly.

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