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An Overview of 30 Years of AREVA TN's Burnup Credit Practices for Transport and Interim Storage of Used Fuel Assemblies

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Abstract

Credit for criticality safety evaluation from the decrease in reactivity during fuel irradiation due to the change in concentration (reduction) of fissile nuclides and the production of neutron absorbing nuclides is commonly referred to as BUrnup Credit (BUC). The use of the BUC approach for transport and dry storage cask applications requires careful considerations of the fuel operating history, validation of calculational methods used for the prediction of isotopic compositions after irradiation for criticality assessments, and also considerations of additional measures in order to ensure safe cask loadings.

Since the early nineteen eighties, AREVA TN has implemented BUC methods for the demonstration of the sub-criticality of transport and interim storage casks loaded with PWR uranium oxide (UO_2) used fuel assemblies. Initially, a simplified BUC method based on the sole consideration of only major actinides and the use of a partial burnup credit was applied. As long as the fuel enrichment of the PWR UO_2 fuel assemblies is sufficiently low, this simplified BUC approach is satisfactory to cover industrial needs without the necessity of designing new casks or of increasing the amount of neutron absorbers in new basket designs. Nevertheless, the ever-increasing enrichment of PWR UO_2 fuel assemblies over the last decade has led to the development and then to the implementation of an advanced BUC methodology including not only actinides but also major fission products. This advanced BUC method enables AREVA TN to limit both the increase in the neutron poison content in new basket designs as well as the increase in the performance of the casks with the existing baskets without physical modifications. A big step forward has recently been taken by AREVA TN who has obtained, for the first time, approvals from French and German Competent Safety Authorities for the application of this advanced BUC approach to initially highly enriched PWR UO_2 used fuel assemblies.

Introduction

As part of the AREVA Group, AREVA TN design, licenses and manufactures casks for the transportation or interim storage of radioactive materials and Used Nuclear Fuel (UNF) assemblies throughout the entire nuclear fuel cycle. Within this framework, AREVA TN also transports UNF from European Nuclear Power Plants (NPPs) to the nuclear recycling plant of AREVA NC in La Hague (France). Since the early 1970s, AREVA TN has carried out more than 6000 transport operations from different NPPs to the La Hague nuclear recycling plant. For this purpose, AREVA TN operates a wide range of transport casks for UNF assemblies of several designs. The use of transport casks is subject to package approvals by the appropriate Competent Safety Authorities. To obtain a package (cask and its content) approval, the cask designer must demonstrate, among other topics, the sub-criticality of the package under the most reactive conditions throughout the transport cycle, including the loading and unloading phases.

Usually, transport casks for Light Water Reactor (LWR) UNF assemblies are designed, from the criticality safety point of view, with the assumption of fresh fuel and with a uniform isotopic composition corresponding to the maximum allowable initial enrichment. Due to the increase of the fuel enrichment since the early nineteen eighties AREVA TN has been progressively implementing BUrnup Credit (BUC) methods for the demonstration of the sub-criticality of transport casks and dual purpose (transport and interim storage) casks loaded with PWR Uranium Oxide (UO_2). Initially, a simplified BUC approach based on the sole consideration of major actinides and the use of a simplified axial burnup was applied. As long as the fuel enrichment of the PWR UO_2 fuel assemblies is sufficiently low, this simplified BUC approach is satisfactory to cover the needs without the necessity of designing new casks or of increasing the amount of neutron absorbers in new basket designs. Nevertheless, the continuous increase of fuel enrichment during the last decade has led AREVA TN to implement an advanced BUC approach. The implementation of this method has the advantage of exploiting the benefits from the negative reactivity reserves, which might be gained by considering not only major actinides but also a limited number of Fission Products (FPs).This improved method also requires the experimental validation of the calculation tools used for both the fuel inventory and the criticality calculations. This is done by using proprietary data under Non-Disclosure Agreements (NDA). The advanced BUC approach was approved for the first time in 2013 by French and German Competent Safety Authorities for the TN® 17/2 transport cask and the TN® 24 E dual purpose cask.

The first part of this paper describes the first steps of AREVA TN's implementation of this BUC methodology for the transportation of UNF applications. Thereafter, the methodology used for the implementation of the advanced BUC method is described including actinides and FPs. The gain in terms of cask performance associated with the improved BUC method is also given through a transport cask and a dual purpose cask as examples. Finally, the new BUC developments in progress at AREVA TN are briefly described in the last section of this paper.

BUC methodology implementation: Outset

Up to the 1980s, criticality safety analyses for transport casks of PWR UO_2UNF were conducted by assuming the fuel to be fresh with uniform isotopic compositions corresponding to the maximum allowable initial enrichment. This assumption led to considerable safety margins. In addition, at that time, the general trend of European NPP_S was to burn more and more reactive fuel elements with higher initial enrichment in the fissile material for higher final burnup. Due to this continuous increase in the initial fuel enrichment, since the early nineteen eighties AREVA TN has been interested in the implementation of a BUC method to improve the capacity of its transport packagings without physical modifications of the existing basket designs. For example, the initial enrichment of German 1300 MWe PWR NPP UNF assemblies has increased from about 3.30 wt.% ^{235}U in the beginning of the 1980s to about 3.55 wt. % in the mid-1980s to now over 4.50 wt. %^{235}U. For this reason, the traditional assumption of considering fresh fuel for criticality safety assessments has proven to be insufficient for the existing basket designs to accommodate the new types of fuel elements to be transported or stored.

The strategy of AREVA TN was to benefit from the negative reactivity gained by considering the main actinides in the criticality safety analyses. Therefore, in the mid-1980s AREVA TN developed a restricted BUC approach based on the consideration of only 8 major actinides (^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu) and a certain amount of burnup corresponding to the 50 least-irradiated centimeters of the active length of the fuel assembly (see Figure 1). Moreover, this BUC approach considered the following very conservative assumptions:

– A uniform (or flat) axial burnup profile corresponding to the 50 least-irradiated centimeters of the active length of the fuel assembly was considered in the criticality analysis.
– The value of the burnup taken in the criticality studies had to be lower than the mean value reached in the 50 least-irradiated centimeters of the active length of each loaded UNF assembly.
– The value of the mean burnup in the 50 least-irradiated centimeters was verified by a measurement or irradiation control (depending on the burnup level retained in the calculations).

This BUC methodology based on the consideration of major actinides is commonly referred to as the “50 least-irradiated centimeters” method.

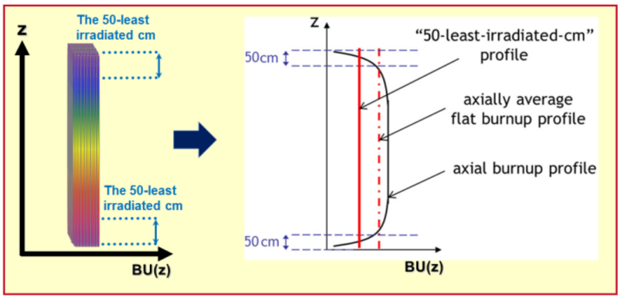

Figure1. The 50 least-irradiated cm profile

Additionally, one of the requirements for criticality analysis with UNF assemblies in a transport or dual purpose (transport and storage) cask was to validate the criticality code used for BUC calculations and associated nuclear data. In the 1980s an experimental French program was conducted to support the development of a technical basis for BUC validation in the case of industrial configurations: transport, storage, recycling of UNF assemblies. A series of critical experiments referred to as the “Haut Taux de Combustion” (HTC) [1] experimental program (see Figure 2), unique in the world (proprietary data under NDA), was conducted at the Valduc research facility (France). These experiments were designed by the Institut de Radioprotection et de Sûreté Nucléaire (IRSN) and co-funded by AREVA NC and the IRSN. The aim of this experimental program was to validate the nuclear data of major actinides and the HTC rods by simulating typical PWR UO_2used fuel with an initial enrichment of 4.50 wt. %^{235}U.

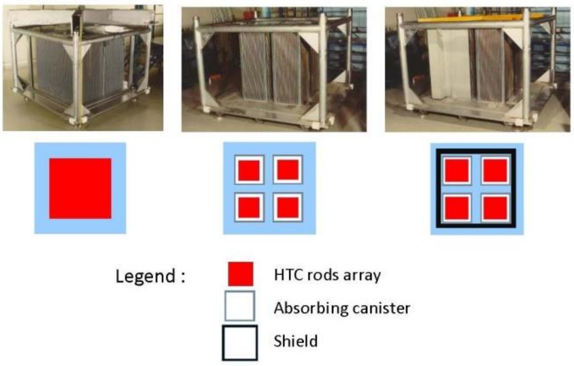

Figure 2. HTC critical experiment configurations

AREVA TN’s implementation of this limited BUC approach for the transportation of PWR UO_2 used fuel assemblies was done in two steps depending on the magnitude of burnup guaranteed on the top and bottom 50-cm end of the active length of the UNF assembly.

- **First step: qualitative (Go/No-Go) burnup verification**

A first BUC approach was accepted for the first time in 1987 by the French Competent Safety Authority and then validated in foreign countries (particularly in Germany) for PWR 16×16 UO_2UNF assemblies loaded in the TN®13 transport cask family. Later equipped with basket type 904 or 924, these were originally designed to load 12 PWR UO_2UNF assemblies. Under the fresh fuel assumption, the capacity of the cask was limited to 11 PWR UO_2 used fuel assemblies having a maximum initial enrichment of 3.31 wt.%^{235}U. A first burnup credit methodology considering a burnup level of 3.20 GWd/MTU was considered in the criticality safety assessment of the TN®13/2 transport cask to extend the cask capacity to an initial enrichment of 3.55 wt.%^{235}U with a loading of 11 PWR UO_2UNF assemblies.

The approval for this first BUC method was obtained from French Competent Safety Authority with the following requirements:

i. Guarantee of a safety margin on the irradiation: The authorized BUC had to be calculated as an average value in the 50 least-irradiated centimeters of the active fuel length instead of an average value over the total active length.

ii. On the basis of the fuel management and in-core measurement, the operator of the power plant had to guarantee that the minimum average burnup in the least irradiated part (i.e. the last 50 cm) of the active fuel length exceeded the allowed burnup level after one cycle, and that each fuel assembly to be loaded in the cask had been irradiated for at least one cycle.

iii. The irradiation status of each fuel assembly had to be checked by a qualitative measurement (irradiation control by gamma scanning only) in the reactor pool before cask loading. These measurements had to be in accordance with the NPP quality assurance program.

iv. In addition, the Competent Safety Authorities required the demonstration of the reliability of the fuel identification procedures.

This first BUC approach became a common practice for AREVA TN and it continues to be used both for the transportation of EDF (Electricté De France) NPP used fuel assemblies to the La Hague recycling plant (TN®17/2,TN®12/2, and TN®13/2 transport cask families) and for European dual purpose casks (TN®24 cask family).

❖ **Second step: quantitative burnup verification**

In 1993 a second step in the application of the BUC approach was made by AREVA TN for the transportation of PWR UO_2 used fuel assemblies with an initial enrichment over 4 wt. %^{235}U and an average burnup level greater than 3.20 GWd/MTU. This was mainly due to the need for of transportation of used fuel assemblies from European NNPs (except EDF NPPs) to the La Hague recycling plant taking into account different designs of PWR UO_2 (15×15, 16×16, or 18×18) types of assemblies. The level of the average burnup used in the criticality safety assessment can reach 20 GWd/MTU. For example, the TN®13/2 transport cask equipped with a new basket design (type 928) was used for the transportation of UNF assemblies from the GROHNDE (Germany) NPP to the La Hague recycling plant. The cask was loaded with its maximum capacity (12 fuel assemblies) by using the BUC approach based on the consideration of 8 main actinides and an average burnup level of 12 GWd/MTU. In the same way, the TN®17/2 transport cask equipped with basket type 903 with a capacity of 7 PWR UO_2 type fuel assemblies was licensed in France and Germany for PWRUO_2 15×15 type used fuel assembly transportation with an initial enrichment of 4.05 wt. %^{235}U:

– 6 UNF assemblies: with the average burnup in the 50 least-irradiated centimeters greater or equal to 3.2 GWd/MTU with qualitative burnup verification (gamma scanning) requirements prior to the loading into the cask,

– 7 UNF assemblies: with the average burnup in the 50 least-irradiated centimeters greater than or equal to 16 GWd/MTU with additional requirements in terms of burnup verification prior to the loading into the cask.

To compensate for this increase in the average burnup, a new requirement was introduced in the BUC approach which consisted of replacing the previous qualitative irradiation go/no-go check by a quantitative measurement of the actual average UNF burnup in the least-irradiated part (i.e. the last 50 cm) of the active fuel length with validated instruments. The acceptance of each fuel assembly was subjected to a comparison of the physical measurements with minimum criteria and a cross check with data supplied by the utilities.

The gain obtained from this limited BUC approach in terms of performance for transportation of UNF assemblies is illustrated in Table 1 for the TN®13/2 transport cask loaded with PWR UO_2 16×16 type fuel assemblies.

Table 1. Current approvals for the TN13/2 transport cask

<table>
<tr><th>Cask</th><th>Number of compartments in the basket</th><th>Number of fuel elements</th><th>Initial enrichment (wt. %^{235}U)</th><th>Minimum average burnup required(*) (GWd/MTU)</th><th>Requirements</th></tr>
<tr><td rowspan="3">TN ® 13/2</td><td rowspan="3">12</td><td>11</td><td>≤ 3.31</td><td>0</td><td>None</td></tr>
<tr><td>11</td><td>≤ 3.55</td><td>3.20</td><td>Qualitative burnup control</td></tr>
<tr><td>12</td><td>≤ 4.00</td><td>12.0</td><td>Quantitative burnup measurement</td></tr>
</table>

(*): corresponding to the 50 least-irradiated centimeters of the fuel assembly

The “50 least-irradiated centimeters” method has been used for more than 25 years by AREVA TN on the TN12, TN13, and TN17 transport cask families for the transportation of PWR UO_2UNF assemblies from European NPPs to the AREVA NC La Hague nuclear recycling plant but also in a number of dual purpose casks (TN®24 cask family). Nevertheless, this method is limited today, particularly due to the increase in the proportion of initially highly enriched used fuel assemblies. Thus, it has become necessary for AREVA TN to improve the previous limited BUC approach so as to limit both the increase in the neutron poison content in new basket designs and the burnup constraints attached to the acceptability of UNF assembly transportation.

Implementation of advanced burnup credit

As the initial enrichment of fuel assemblies has increased from 4 wt. %^{235}U to 5 wt. %^{235}U, it has become necessary to use less conservative assumptions for transportation and interim storage of UNF assemblies than those based on very conservative assumptions of the “50-least-irradiated-centimeters” BUC method. The strategy adopted by AREVA TN consists in taking benefits from the negative reactivity worth obtained from some Fission Products (FPs) and a more realistic axial burnup profile in criticality safety analysis.

In 2013 AREVA TN licensed for the first time the TN®17/2 transport cask (France) and TN®24E dual purpose cask (Germany) with an advanced BUC method which takes into account 9 actinides (^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{241}Am) and 6 FPs (^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{152}Sm, and ^{155}Gd) in criticality assessments. This improved BUC method was performed in accordance with the research programs carried out by the French CBU working group [2], and it dealt, in particular, with the following requirements:

- the establishment of conservative conditions of irradiation to guarantee the conservatism of the UNF calculation,
- the use of bounding axial burnup profiles for the UNF assemblies that could be loaded into the cask,
- the experimental validation of both the UNF compositions (depletion codes) and the criticality calculations,
- the axial burnup and burnup verification requirements prior to each loading of PWR UO_2UNF assemblies into the cask.

Concerning the first point, sensitivity studies were conducted on each relevant irradiation parameter (specific power, fuel temperature, temperature and density of the moderator, soluble boron concentration in the moderator…) to define a conservative approach for depletion calculations of PWR UO_2 fuel assemblies. In addition, control rods and the depleted fuel environment (typically MOX fuel assemblies when necessary) were taken into account in the definition of the advanced AREVA TN BUC approach.

As the burnup distribution in the fuel assembly can also have a significant impact on criticality safety, it needed to be reassessed. Indeed, the criticality of the irradiated fuel assemblies is affected by their axial burnup shapes. Generally, for transportation of UNF assembly applications, a partial burnup (< 20 GWd/MTU) of the total average burnup of the fuel assemblies is considered to cover the needs in the criticality safety analysis. Moreover, it is useful to have a simple and practical method to define a bounding axial burnup profile to cover the transportation or the interim storage of several types of PWR UO_2UNF assembly designs with different irradiation conditions.

In this sense, a generic method for the definition of bounding uniform axial burnup profile was developed by AREVA TN in accordance with the international BUC standard ISO 27468 [3]. The axial burnup profile considered in the criticality safety evaluation came from the determination of the most limiting axial burnup profile among calculated profiles from in-core measurements [4], measured profiles from the La Hague recycling plant from different European PWR UO_2 used fuel assembly designs, and asymmetrical axial burnup profiles [5]. The penalizing profile was given by the definition of a ratio between the average burnup of the fuel assemblies to be loaded in the cask and the average burnup in their 50 least-irradiated centimeters corresponding to the level of burnup used in the criticality analysis. This approach had already been validated by the French Competent Safety

Authority (ASN) for the implementation of the advanced BUC methodology on the TN®17/2 cask loaded with 7 PWR UO_2 17×17 type fuel assemblies initially enriched at 4.60 wt. %^{235}U.

An alternative to the previous generic method to determine bounding axial burnup profiles for criticality safety analyses was be obtained by limiting the fuel assemblies to be loaded into the cask to only a few NPPs. Hence, it was possible to calculate different types of profiles based on the reactor records of these specific NPPs and then to identify a penalizing one. However, the conservatism of the method used for determining the bounding axial profile had to be demonstrated. This alternative approach was used by AREVA TN for the approval of the TN®24E dual purpose cask for the German markets which can be loaded with 21 PWR UO_2 used fuel assemblies initially enriched at 4.65 wt. %. Bounding axial burnup profiles was generated by AREVA NP on the basis of representative axial fuel assembly burnup shapes coming from the German NPPs derived from in-core 3D power density distribution measurements.[6]

The experimental validation of the calculation tools (depletion and criticality codes) was done by using different experimental programs conducted to support the implementation of the advanced BUC method. Indeed, a BUC program [7] was developed at Cadarache Centre (France) in the framework of a CEA (Commissariat à l'Energie Atomique et aux Energies Alternatives)-AREVA collaboration to validate the UNF inventory calculations. The French Post Irradiation Examination (PIE) proprietary data, which included chemical analyses and microprobe measurements of PWR fuel rods, were used for validation of the UNF inventory. The French PIE data were based on chemical analysis measurements from fuel rod cuts irradiated in French PWR reactors. This covered a large range of UO_2 fuel with various levels of enrichment in ^{235}U, from 3.10 wt. %^{235}U to 4.50 wt. %^{235}U, and burnups from 10 GWd/MTU to 60 GWd/MTU. The French PIEs provided high quality experimental measurements (“clean experiments”) of the isotopic inventory of irradiated fuel samples with very low experimental uncertainties due, in particular, to the accurate knowledge of the irradiation history in the EDF reactors. Furthermore, the French PIE data were reviewed and compared to PIE results from international experiments, available through public data, the French (ASN), and the German (Bfs) Competent Safety Authorities. The latter confirmed the accuracy of French PIE data for the advanced BUC implementation. Regarding the experimental validation of the criticality computer code system, a French program was also performed to support the development of a technical basis for the implementation of BUC for industrial configurations (transportation, storage…). A set of “Haut Taux de Combustion” (previously described) and Fission Products experiments [8], co-funded by AREVA NC and the IRSN, were conducted at the Valduc research facility (France). The French HTC and FPs experiments were conducted to validate the cross sections of major actinides (^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, and ^{241}Am) and main 6 FPs (^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{152}Sm, and ^{155}Gd) for a representative range of fuel cycle configurations. These experiments were suitable for use in the validation of criticality models of transport or dual purpose casks with the use of the advanced BUC methodology.[9] The evaluation of the French HTC experiments has shown low experimental uncertainties,[10] proving that these experiments were planned and performed under the most rigorous conditions. In addition, a good agreement between the French HTC and FPs experiments and the calculation for BUC applications was obtained showing no significant bias with regard to experimental uncertainties.

Finally, among the main requirements associated with the implementation of the advanced BUC, a burnup verification procedure was executed prior to each loading of PWR UO_2UNF assemblies into the transport or dual purpose cask. This procedure ensured that each PWR UO_2UNF assembly to be loaded met the conditions of the minimum required average burnup as well as the discharge axial burnup profile bounded by the corresponding bounding axial burnup profiles used in the criticality safety assessment. The establishment of such a procedure now enables consignors to safely load UNF assemblies into the cask and guarantees the compliance of such a loading with the national and international criticality safety requirements [11].

Approvals for the implementation of the advanced BUC method including actinides and fission products was obtained in 2013 for the TN®17/2 transport cask from French Competent Safety Authority and for the TN®24E dual purpose cask from the German Competent Safety Authority. The TN®17/2 cask can be loaded with PWR UO_2 UNF assemblies initially enriched at 4.60 wt. %^{235}U with an average burnup of 10 GWd/MTU in the criticality calculations. The TN®24E dual purpose cask can be loaded with 21 PWR UO_2 used assemblies and the fuel has a maximum initial enrichment of about 4.60 wt.%^{235}U with an average burnup requirement of 12 GWd/MTU in the criticality calculations. These approvals were obtained with the following requirements:

i. Prior to each loading, in accordance with the NPP quality assurance policy, verification had to be done to ensure that the average burnup of each selected PWR UO_2 used fuel assembly met the minimum required average burnup.

ii. Before loadings, a qualitative irradiation check (gamma dose rate measurement for example) of the PWR UO_2 used fuel assemblies had to be conducted to ensure that fuel assemblies were burned at the core, mainly to avoid any mistake in loading a fresh fuel assembly into the cask.

Additionally, an in-depth defense (events of defense level 2) approach was considered by evaluating an event of misloading one fuel assembly into the TN®17/2 transport cask and TN®24 E dual purpose cask. Such a misloading risk cannot be totally excluded, even if different administrative and technical actions are taken into account by the consignors during loading process. Therefore, the sub-criticality of the packages in such a situation is ensured.

Benefits of the implementation of the advanced burnup credit

Figure 3 illustrates the gain obtained from the advanced BUC approach in the performance of the transportation of UNF assemblies (initial enrichment of the fuel assemblies) compared to the fresh fuel assumption and the limited BUC approach based on the 50 least-irradiated centimeters of the active fuel length of the fuel assembly. These comparisons were made on the TN®17/2 transport (under normal situation of transport) and the TN®24 E dual purpose (under accidental situation of transport) casks loaded respectively with 7 PWR UO_2 17×17 type and 21 PWR UO_2 18×18 type fuel assemblies.

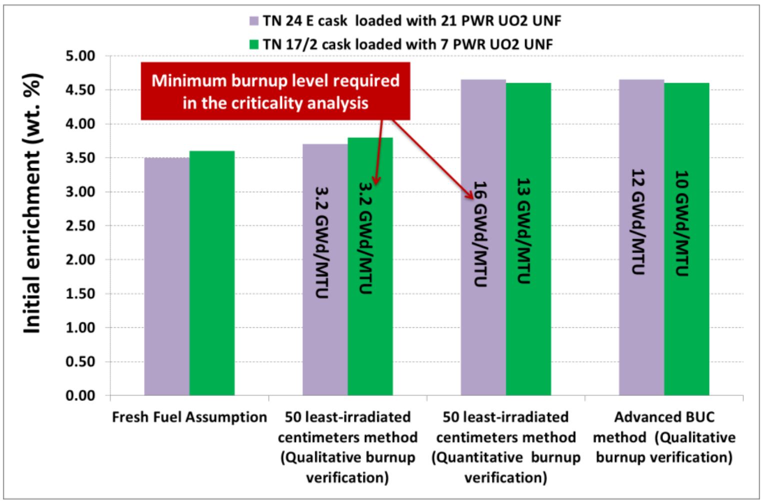

Figure 3. Comparison between fresh fuel assumption and different BUC methods

The advanced BUC method based on the consideration of main actinides and FPs, implemented in a conservative manner, is adequate for highly-enriched PWR UO_2 fuel assembly loading, in particular, concerning the burnup constraints attached to the acceptability of the fuel assemblies for transport or dual purpose casks.

New developments in BUC methodologies

The feedback received from the implementation of the improved BUC approach has led AREVA TN to new perspectives for further cask developments by extending the application area of the advanced PWR UO_2 BUC method to PWR Mixed OXide (MOX) and BWRUO_2 UNF assemblies.

The continuous increase of the plutonium content in PWR MOX fuel assemblies has led AREVA TN to develop the BUC MOX methodology. One of the main challenges of implementing BUC for PWR MOX fuel is the wide range of parameters in comparison to PWR UO_2 fuel such as the quantity of the content, the plutonium isotopic composition, the initial uranium composition, the zoning within the assembly… In addition, due to the change of the plutonium during the irradiation phases in the core of the reactor, the bounding plutonium composition considered under the fresh fuel assumption does not necessarily lead to the most reactive fuel after irradiation. Therefore, even if the definition of a conservative approach using the BUC method for PWR MOX assemblies requires the same analysis as that of the advanced BUC developed for PWR UO_2 assemblies, careful analysis must to be made to ensure conservative assumptions of the spent fuel inventory and criticality calculations.

Regarding BWR UO_2 fuel assembly transportation and interim storage, the AREVA TN strategy consists of taking credit for the presence of Integral Burnable Absorbers (IBA) in the BWR UO_2 fuel assembly designs. Typically, BWR fuel assemblies make heavy use of gadolinium poisoning (Gd_2O_3)

in some fuel rods such as IBA to control the reactivity excess at the beginning of the fuel life in the core. The approach of taking credit from the presence of gadolinium in some fuel rods in the assembly in the criticality safety analysis is commonly referred to as the “gadolinium peak reactivity.” Taking credit of the presence of gadolinium in the fuel assembly design is very often not considered as it is in BUC. However, the gadolinium depletion reactivity peak, which corresponds to the consumption of nearly all of the gadolinium content, is determined as a function of burnup. Therefore, taking advantage of the credit from burnable gadolinium in some fuel rods is therefore a BUC approach. This necessitates the definition of a method to ensure conservative assumptions of the UNF inventory and criticality calculations as is done for the PWR UO_2 assembly type. The evaluation of the use of the gadolinium peak reactivity method depends on the BWR UO_2 fuel assembly design parameters and the core operating conditions. Indeed, BWR UO_2 fuel designs are highly heterogeneous, both radially and axially, with the presence of partial-length fuel rods, each assembly having its specific initial fissile enrichment distribution and gadolinium fuel rod configuration. Depletion calculations of BWR UO_2 fuel assemblies also depend on the local core conditions: mainly the coolant void fraction, the presence of control rods during the operating cycle, the distribution of fissile content in the fuel assemblies, the number of burnable neutron poison fuel rods per fuel assembly, and the positions of these rods in the fuel assemblies.

AREVA TN has developed PWR MOX and BWR UO_2 (gadolinium credit) BUC methods which are in the process of implementation for transport and interim storage cask designs in the near future.

CONCLUSION

The implementation of the BUC approach by AREVA TN for transportation and interim storage of PWR UO_2 UNF assemblies has been done progressively. Since the beginning of the first limited BUC implementation in 1987 to the acceptance by different Competent Safety Authorities (French and German) of the advanced BUC approach in 2013 for the first time, AREVA TN has made a continuous effort to establish and implement BUC methods to improve the cask performance.

The advanced BUC method based on the consideration of actinides and fission products represents a big step limiting the burnup constraints linked to the acceptability of the fuel assemblies for transportation. In addition, this new BUC method may extend burnup credit advantages to new transport and interim storage cask designs dedicated to PWR UO_2UNF assemblies by limiting the increase of the neutron poison content in the new basket designs. The improved BUC method is particularly suitable for the transportation and interim storage of highly enriched PWR UO_2 used fuel assemblies with an appropriate level of criticality safety.

The calculation tools associated with the advanced BUC methodology are validated on a large and suitable experimental database. The validation of the computer code system was benchmarked against French Post Irradiation Examination (PIE) data, and critical experiments (French HTC & FPs) representative of transport and interim storage cask configurations, both of which are proprietary data under NDA.

The use of the BUC approach is now a common practice in France. The different approvals obtained by AREVA TN for the implementation of BUC have been limited to PWR UO_2 used fuel assemblies. New developments regarding BUC methods for PWR MOX and BWR UO_2 (gadolinium credit) are in process of implementation for transport and interim storage cask designs in the near future.

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